PART III

Perspectives and Uses

8. PERSPECTIVES ON FREQUENCY OF CORE DAMAGE

8.1 Introduction

Chapters 3 through 7 have summarized the core damage frequencies individually for the five plants assessed in this study. Significant differences among the plants can be seen in the results, both in terms of the core damage frequencies and the particular events that contribute most to those frequencies. These differences are due to plant-specific differences in the plant designs and operational practices. Despite the plant-specific nature of the study, it is possible to obtain important perspectives that may have implications for a larger number of plants and also to describe the types of plant-specific features that are likely to be important at other plants. This chapter provides some of these perspectives.

8.2 Summary of Results

As discussed in Chapter 2, the core damage frequency is not a value that can be calculated with absolute certainty and thus is best characterized by a probability distribution. It is therefore discussed in this report in terms of the mean, median, and various percentile values. The internal-event core damage frequencies are illustrated graphically in Figure 8.1 (Refs. 8.1 through 8.5). The figure does not include the contributions of external events, which are discussed in Section 8.4.

In Figure 8.1 the lower and upper extremities of the bars represent the 5th and 95th percentiles of the distributions, with the mean and median of each distribution also shown. Thus, the bars include the central 90 percent of the distributions (it should be remembered that the distributions are not uniform within these bars). These figures show that the range between the 5th and 95th percentiles covers from one to two orders of magnitude for the five plants. There is also significant overlap among the distributions, as discussed below. The reader should refer to References 8.1 through 8.5 for detailed discussion of the distributions.

Figures 8.2 and 8.3 show the contributions of the principal types of accidents to the mean core damage frequency for each plant. Figure 8.4 also presents this breakdown, but on a relative scale. These figures show that some types of accidents, such as station blackouts, contribute to the core damage frequencies for all the plants; however, there is substantial plant-to-plant variability among important accident sequences.

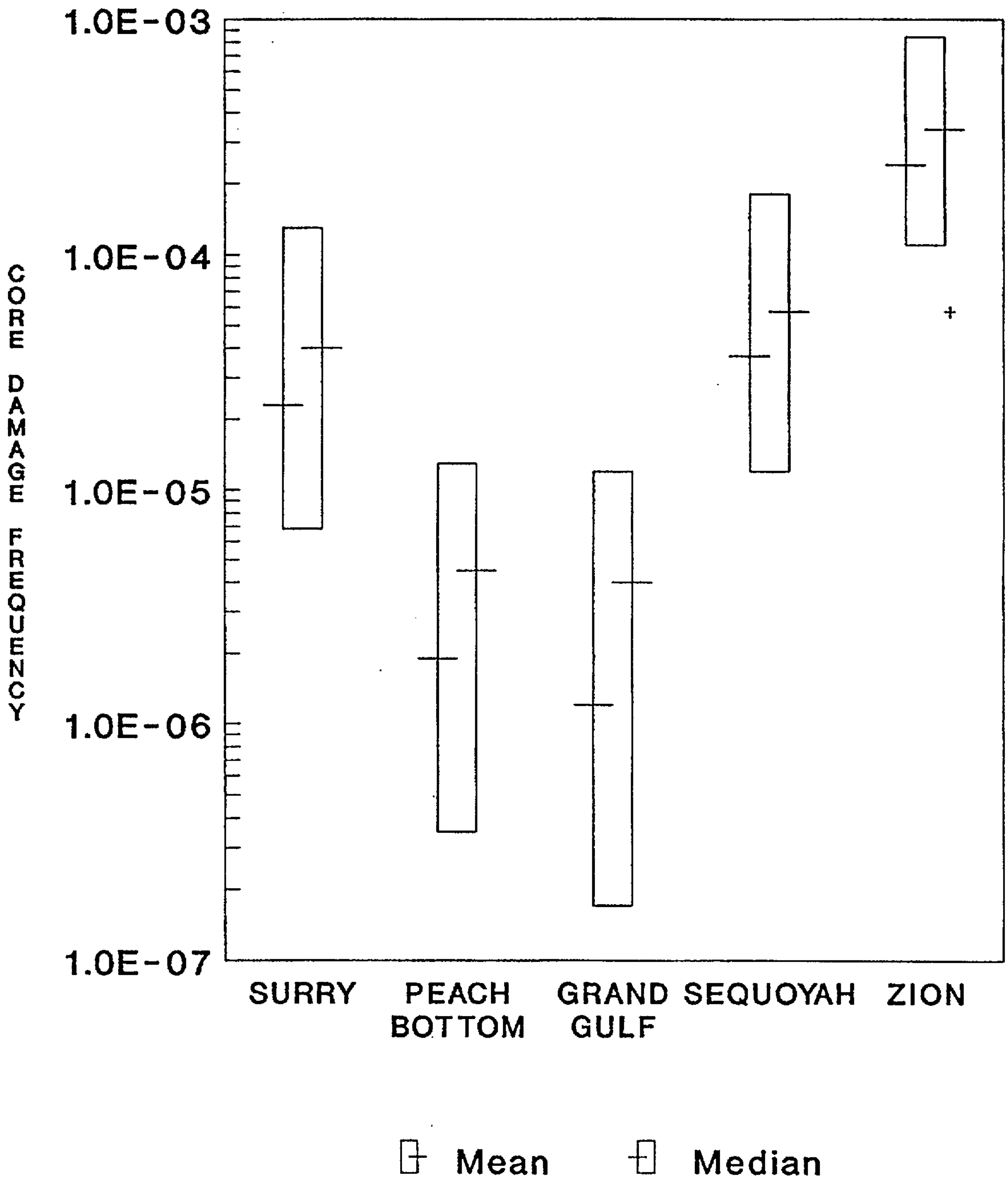
Figures 8.5 through 8.8 provide the results of the external-event analyses, and Figures 8.9 through 8.12 give the breakdown of these analyses according to the principal types of accident sequences.

8.3 Comparison with Reactor Safety Study

Figures 8.13 and 8.14 show the internal core damage frequency distributions calculated in this present study for Surry and Peach Bottom along with distributions synthesized from the Reactor Safety Study (Ref. 8.6), which also analyzed Surry and Peach Bottom. The Reactor Safety Study presented results in terms of medians but not means. It can be seen that the medians are lower in the present work, although observation of the overlap of the ranges shows that the change is more significant for Peach Bottom than for Surry.

There are two important reasons for the differences between the new figures and those of the Reactor Safety Study. The first is the fact that probabilistic risk analyses (PRAs) are snapshots in time. In these cases, the snapshots are taken about 15 years apart. Both plants have implemented hardware modifications and procedural improvements with the stated purpose of increasing safety, which drives core damage frequencies downward.

The second reason is that the state of the art in applying probabilistic analysis in nuclear power plant applications has advanced significantly since the Reactor Safety Study was performed. Computational techniques are now more sophisticated, computing power has increased enormously, and consequently the level of detail in modeling has increased. In some cases, these new methods have reduced or eliminated previous analytical conservatisms. However, new types of failures have also been discovered. For example, the years of experience with probabilistic analyses and plant operation have uncovered the reactor coolant pump seal failure scenario as well as intersystem dependencies, common-mode failure mechanisms, and other items that were less well recognized at the time of the Reactor Safety Study. Of course, this same experience has also uncovered new ways in which recovery can be achieved during the course of a possible core damage scenario (except for the



Notes: As discussed in Reference 8.7, core damage frequencies below 1E–5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

"+" indicates recalculated Zion mean core damage frequency based on recent plant modifications (see Section 7.2.1).

Figure 8.1 Internal core damage frequency ranges (5th to 95th percentiles).

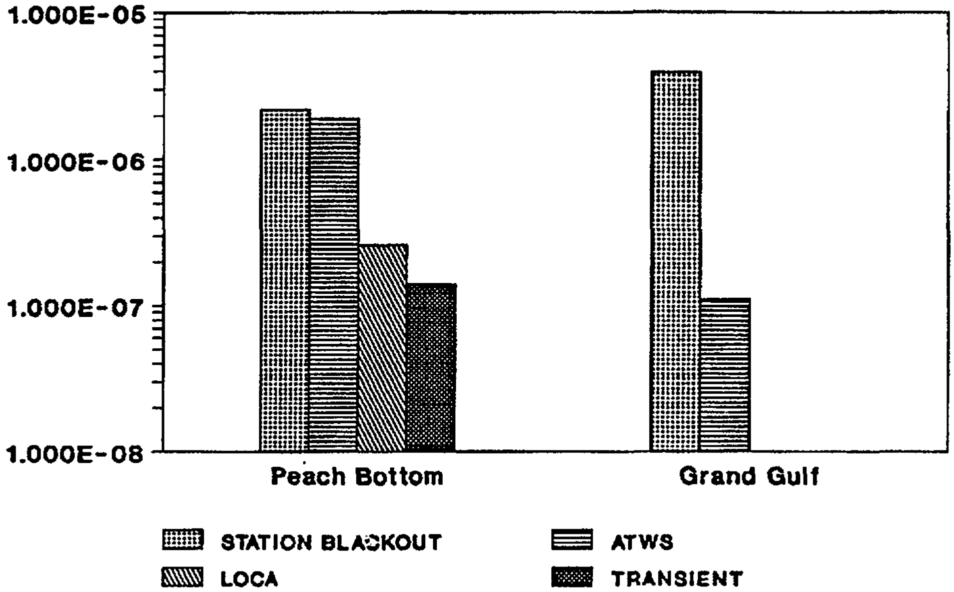
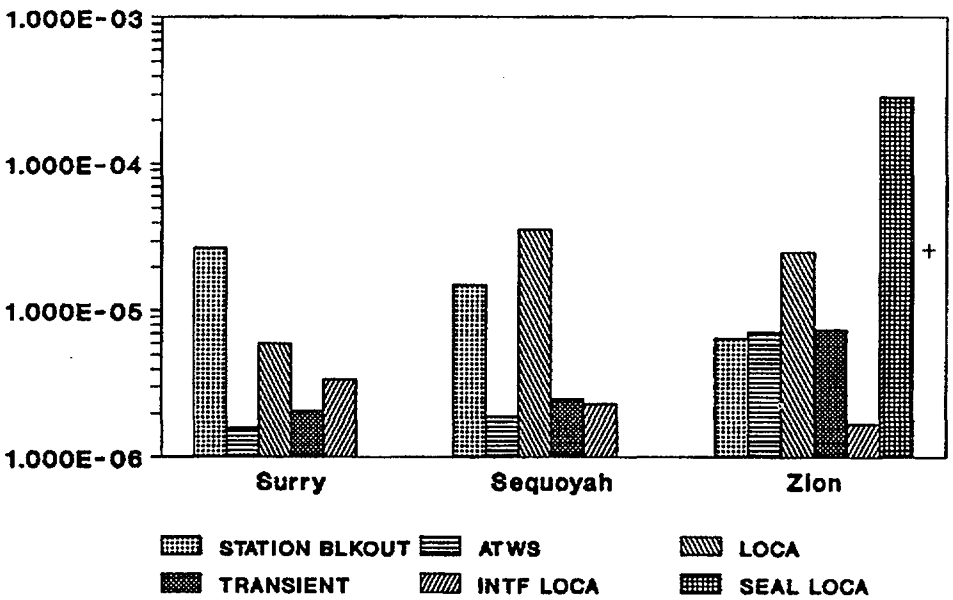


Figure 8.2 BWR principal contributors to internal core damage frequencies.



Notes: As discussed in Reference 8.7, core damage frequencies below 1E–5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

"+" indicates recalculated mean seal LOCA plant damage state frequency based on recent plant modifications (see Section 7.2.1).

Figure 8.3 PWR principal contributors to internal core damage frequencies.

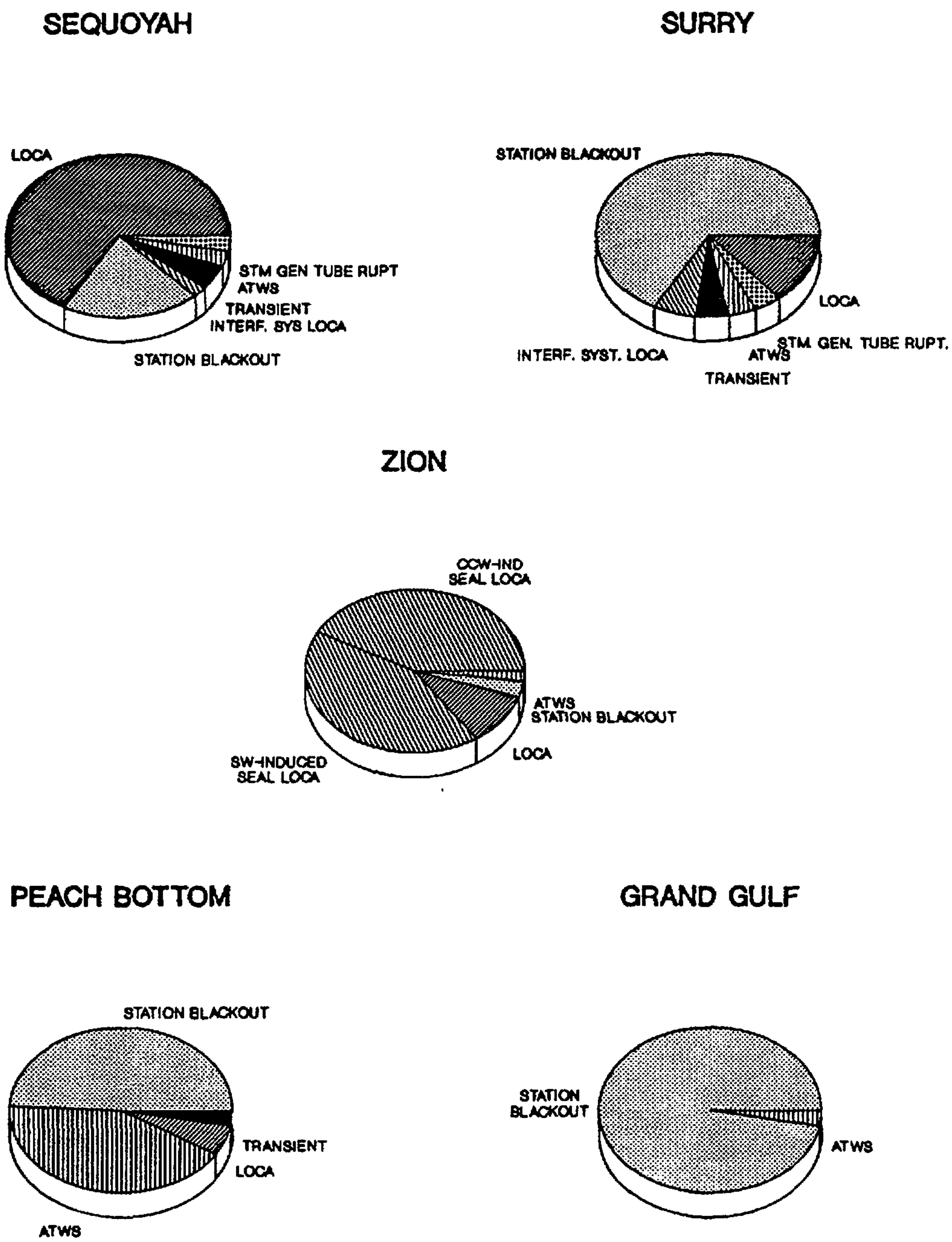


Figure 8.4 Principal contributors to internal core damage frequencies.

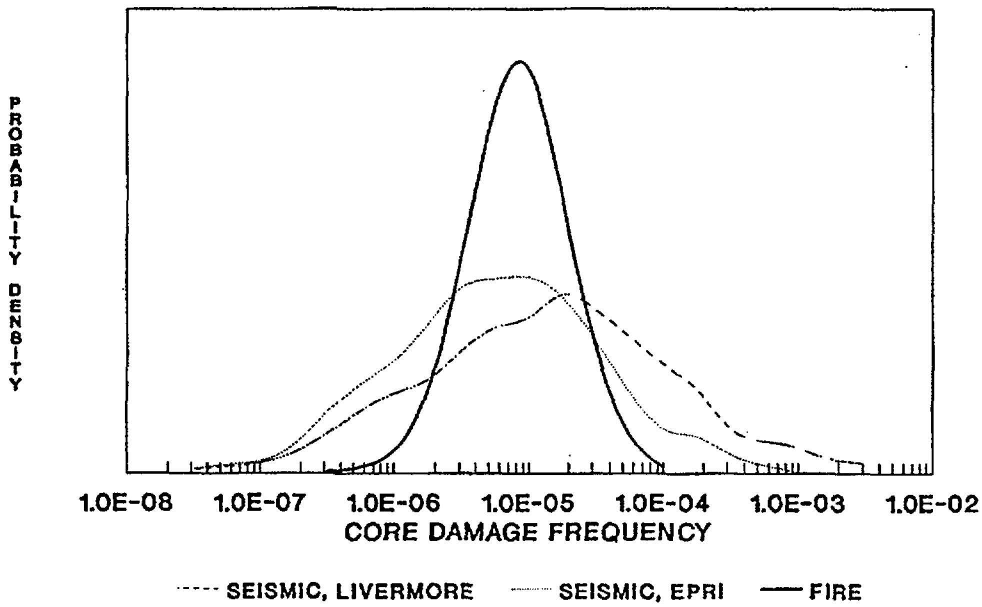
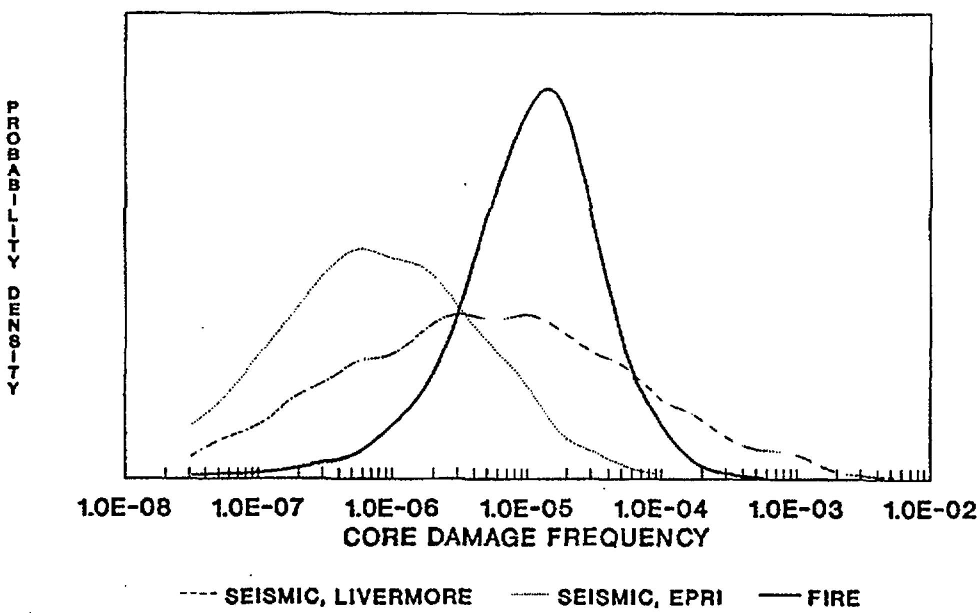


Figure 8.5 Surry external-event core damage frequency distributions.



Note: As discussed in Reference 8.7, core damage frequencies below 1E–5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Figure 8.6 Peach Bottom external-event core damage frequency distributions.

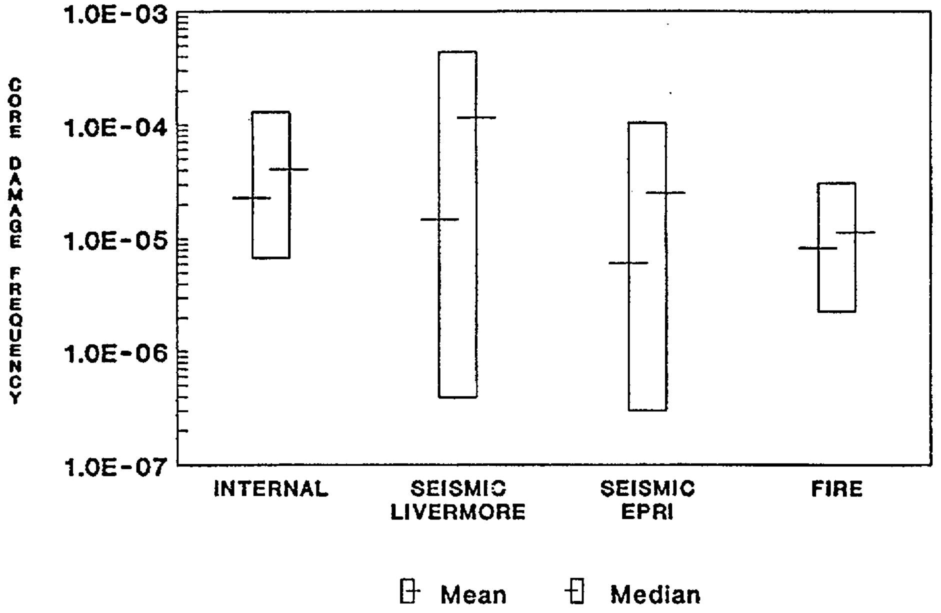
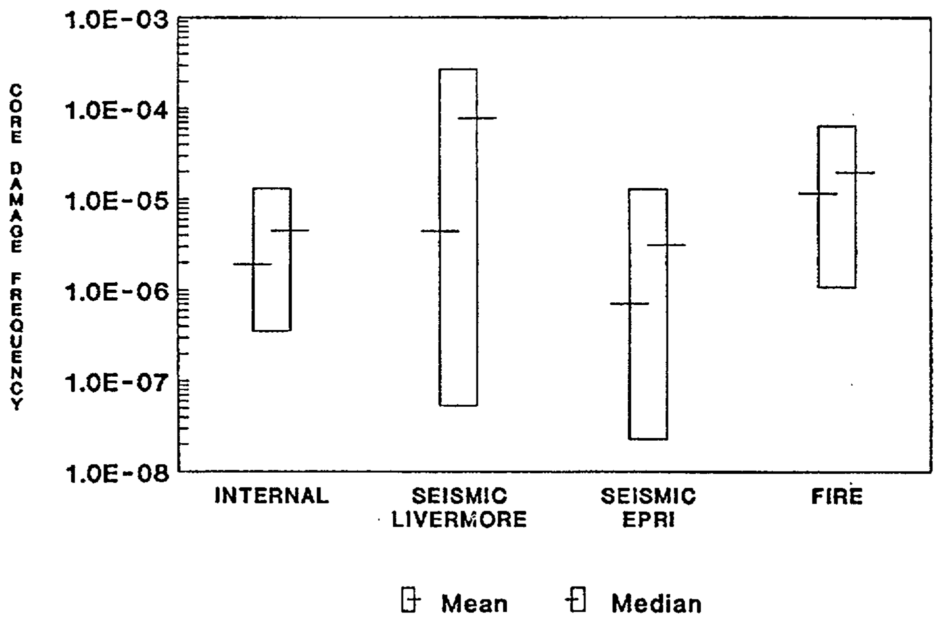


Figure 8.7 Surry internal- and external-event core damage frequency ranges.



Note: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Figure 8.8 Peach Bottom internal- and external-event core damage frequency ranges.

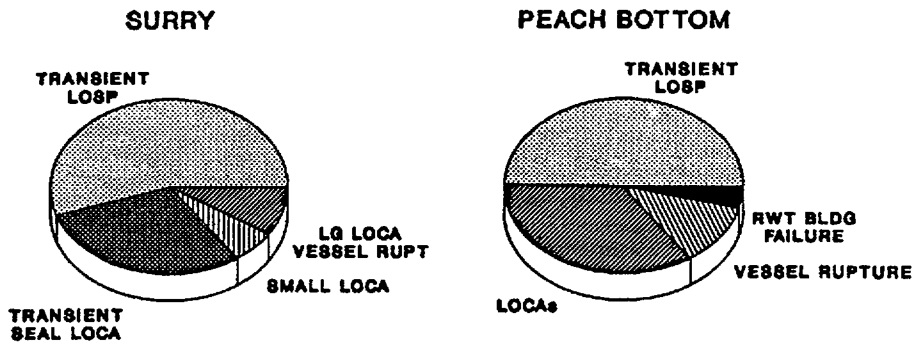


Figure 8.9 Principal contributors to seismic core damage frequencies.

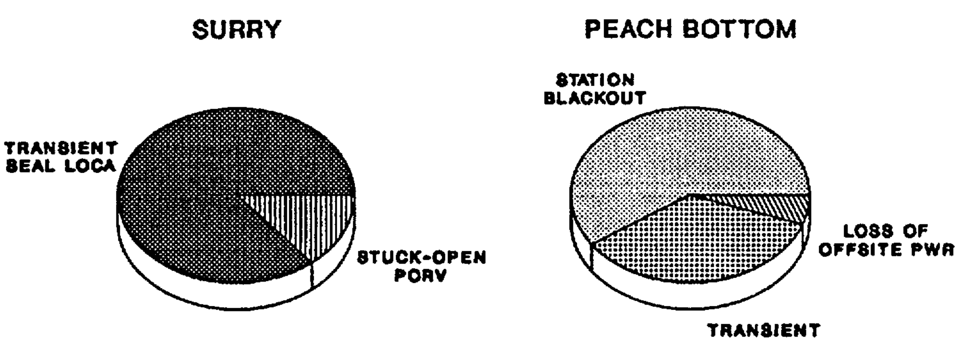


Figure 8.10 Principal contributors to fire core damage frequencies.

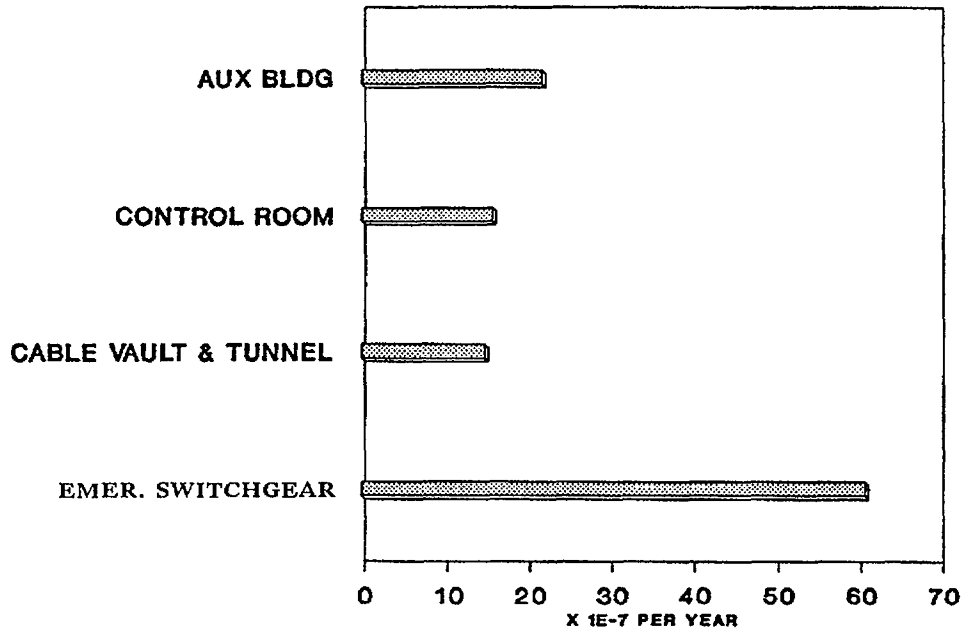
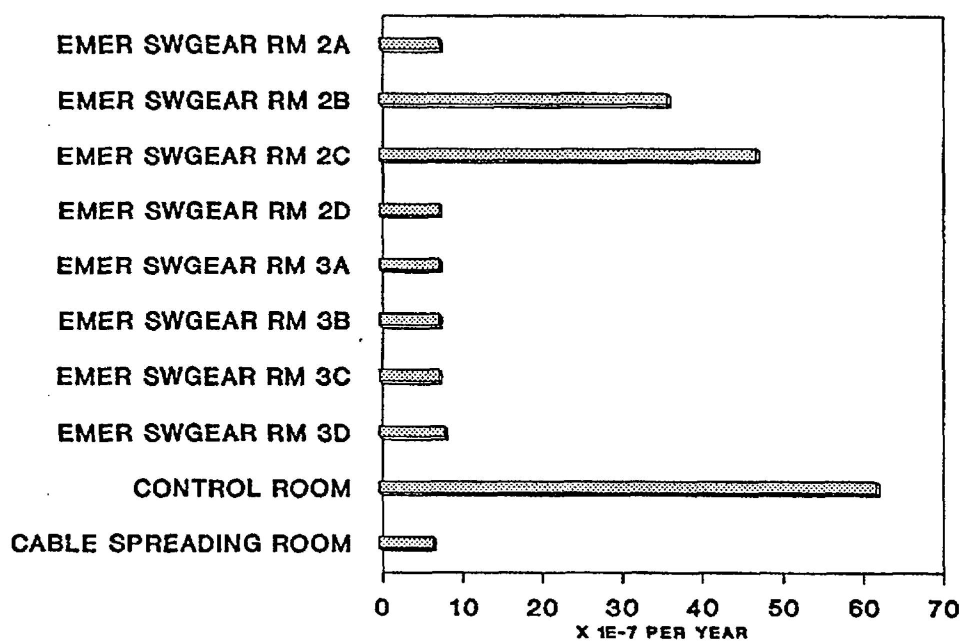


Figure 8.11 Surry mean fire core damage frequency by fire area.



Note: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Figure 8.12 Peach Bottom mean fire core damage frequency by fire area.

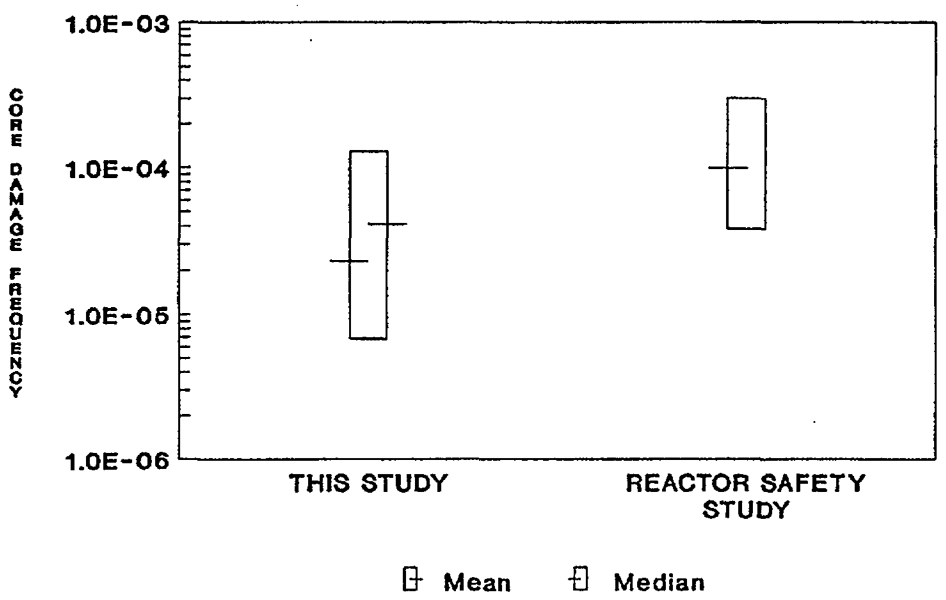
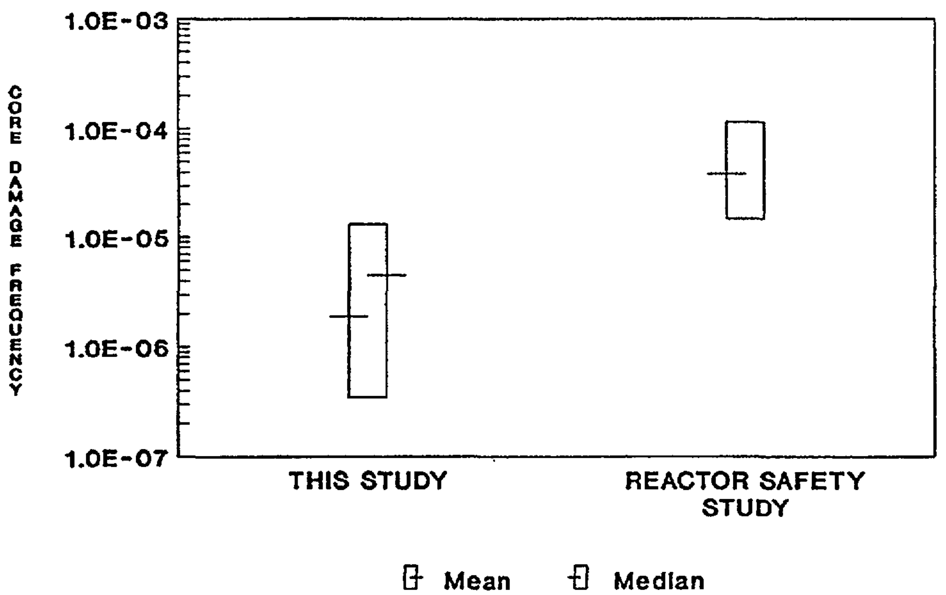


Figure 8.13 Comparison of Surry internal core damage frequency with Reactor Safety Study.



Note: As discussed in Reference 8.7, core damage frequencies below 1E–5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Figure 8.14 Comparison of Peach Bottom internal core damage frequency with Reactor Safety Study.

recovery of ac power, the Reactor Safety Study did not consider recovery actions). Thus, the net effect of including these new techniques and experience is plant specific and can shift core damage frequencies in either higher or lower directions.

In the case of the Surry analysis, the Reactor Safety Study found the core damage frequency to be dominated by loss-of-coolant accidents (LOCAs). For the present study, station blackout accidents are dominant, while the LOCA-induced core damage frequency is substantially reduced from that of the Reactor Safety Study, particularly for the small LOCA events. This occurred in spite of a tenfold increase in the small LOCA initiating event frequency estimates, which was a result of the inclusion of reactor coolant pump seal failures. One reason for the reduction lies in plant modifications made since the Reactor Safety Study was completed. These modifications allow for the crossconnection of the high-pressure safety injection systems, auxiliary feedwater systems, and refueling water storage tanks between the two units at the Surry site. These crossties provide a reliable alternative for recovery of system failures. Thus, the plant modifications (the crossconnections) have driven the core damage frequencies downward, but new PRA information (the higher small LOCA frequency) has driven them upward. In this case, the net effect is an overall reduction in the core damage frequency for internal events.

In the case of Peach Bottom, the Reactor Safety Study found the core damage frequency to be comprised primarily of ATWS accident sequences and of transients with long-term failure of decay heat removal. The present study concludes that station blackout scenarios are dominant. The possibility of containment venting and allowing for some probability of core cooling after containment failure has considerably reduced the significance of the long-term loss of decay heat removal accidents. In addition, the plant has implemented some ATWS improvements, although ATWS events remain among the dominant accident sequence types. Moreover, more modern neutronic and thermal-hydraulic simulations of the ATWS sequences have calculated lower core power levels during the event, allowing more opportunity for mitigation such as through the use of low-pressure injection systems. Thus, for Peach Bottom, both advances in PRA methodology and plant modifications have contributed to a reduction in the estimated core damage frequency from internal events.

In summary, there have been reductions in the core damage frequencies for both plants since the Reactor Safety Study. The reduction in core damage frequency for Peach Bottom is more significant than for Surry; however, there is still considerable overlap of the uncertainty ranges of the two studies. The conclusion to be drawn is that the hardware and procedural changes made since the Reactor Safety Study appear to have reduced the core damage frequency at these two plants, even when accounting for more accurate failure data and reflecting new sequences not identified in the Reactor Safety Study (e.g., the reactor coolant pump seal LOCA).

8.4 Perspectives

8.4.1 Internal-Event Core Damage Probability Distributions

The core damage frequencies produced by all PRAs inherently have large uncertainties. Therefore, comparisons of frequencies between PRAs or with absolute limits or goals are not simply a matter of comparing two numbers. It is more appropriate to observe how much of the probability distribution lies below a given point, which translates into a measure of the probability that the point has not been exceeded. For example, if the median were exactly equal to the point in question, half of the distribution would lie above and half below the point, and there would be a 50 percent probability that the point had not been exceeded.

Similarly, when comparing core damage frequencies calculated for two or more plants, it is not sufficient to simply compare the mean values of the probability distributions. Instead, one must compare the entire distribution. If one plant's distribution were almost entirely below that of another, then there would be a high probability that the first plant had a lower core damage frequency than the second. Seldom is this the case, however. Usually, the distributions have considerable overlap, and the probability that one plant has a higher or lower core damage frequency than another must be calculated. References 8.1 through 8.5 contain more detailed information on the distributions that would support such calculations.

Although the distributions are not compared in detail here, the overlap of such core damage frequency distributions is clearly shown in Figure 8.1. For example, one can have relatively high confidence that the internal-event core damage frequency for Grand Gulf is lower than that of Sequoyah or Surry. Conversely, it can readily be seen that the differences in core damage frequency between Surry and Sequoyah are not very significant.

Interpretation of extremely low median or mean core damage frequencies (<1E-5) is somewhat difficult. As discussed in Section 1.3 and in Reference 8.7, there are limitations in the scope of the study that could lead to actual core damage frequencies higher than those estimated. In addition, the uncertainties in the sequences included in the study tend to become more important on a relative scale as the frequency decreases. A very low core damage frequency is evident for Grand Gulf with the median of the distribution in the range of 1E-6 per reactor year. However, it is incomplete to simply state that the core damage frequency for this plant is that low since the 95th percentile exceeds 1E-5 per reactor year. Thus, although the central tendency of the calculation is very low, there is still a finite probability of a higher core damage frequency, particularly when considering that the scope of the study does not include certain types of accidents as discussed in Section 1.3.

8.4.2 Principal Contributors to Uncertainty in Core Damage Frequency

In Section 8.4.3, analyses are discussed concerning some of the issues and events that contribute to the magnitude of the core damage frequency. Generally, for the accident frequency analysis, the issues that contribute most to the magnitude of the frequency are also the issues that contribute most to the estimated uncertainty. More detail concerning the contributions of various parameters to the uncertainty in core damage frequency may be found in References 8.1 through 8.5. Perspectives on the contributions of accident frequency issues to the uncertainty in risk may be found in Chapter 12.

8.4.3 Dominant Accident Sequence Types

The various accident sequences that contribute to the total core damage frequency can be grouped by common factors into categories. Older PRAs generally did this in terms of the initiating event, e.g., transient, small LOCA, large LOCA. Current practice also uses categories, such as ATWS, seal LOCA, and station blackout. Generally, these categories are not equal contributors to the total core damage frequency. In practice, four or five sequence categories, sometimes fewer, usually contribute almost all the core damage frequency. These will be referred to below as the dominant plant damage states (PDSs).

It should be noted that the selection of categories is not unique in a mathematical sense, but instead is a convenient way to group the results. If the core damage frequency is to be changed, changing something common to the dominant PDS will have the most effect. Thus, if a particular plant had a relatively high core damage frequency and a particular group of sequences were high, a valuable insight into that plant's safety profile would be obtained.

It should also be noted that the importance of the highest frequency accident sequences should be considered in relationship to the total core damage frequency. The existence of a highly dominant accident sequence or PDS does not of itself imply that a safety problem exists. For example, if a plant already had an extremely low estimated core damage frequency, the existence of a single, dominant PDS would have little significance. Similarly, if a plant were modified such that the dominant PDS were eliminated entirely, the next highest PDS would become the most dominant contributor.

Nevertheless, it is the study of the dominant PDS and the important failures that contribute to those sequences that provides understanding of why the core damage frequency is high or low relative to other plants and desired goals. This qualitative understanding of the core damage frequency is necessary to make practical use of the PRA results and improve the plants, if necessary.

Given this background, the dominant PDSs for the five studies are illustrated in Figures 8.2, 8.3, and 8.4. Additional discussion of these PDSs can be found in Chapters 3 through 7. Several observations on these PDSs and their effects on the core damage frequency can be made, as discussed below.

Boiling Water Reactor versus Pressurized Water Reactor

It is evident from Figure 8.1 that the two particular BWRs in this study have internal-event core damage frequency distributions that are substantially lower than those of the three PWRs. While it would be inappropriate to conclude that all BWRs have lower core damage frequencies than PWRs, it is useful to consider why the core damage frequencies are lower for these particular BWRs.

The LOCA sequences, often dominant in the PWR core damage frequencies, are minor contributors in the case of the BWRs. This is not surprising in view of the fact that most BWRs have many more systems than PWRs for injecting water directly into the reactor coolant system to provide makeup. For BWRs, this includes two low-pressure emergency core cooling (ECC) systems (low-pressure coolant injection and low-pressure core spray), each of which is multitrain; two high-pressure injection systems (reactor core isolation cooling and either high-pressure coolant injection or high-pressure core spray); and usually several other alternative injection systems, such as the control rod drive hydraulic system, condensate, service water, firewater, etc. In contrast, PWRs generally have one high-pressure and one low-pressure ECC system (both multitrain), plus a set of accumulators. The PWR ECCS does have considerable redundancy, but not as much as that of most BWRs.

For many types of transient events, the above arguments also hold. BWRs tend to have more systems that can provide decay heat removal than PWRs. For transient events that lead to loss of water inventory due to stuck-open relief valves or primary system leakage, BWRs have numerous systems to provide makeup. ATWS events and station blackout events, as discussed below, affect both PWRs and BWRs.

BWRs have historically been considered more subject than PWRs to ATWS events. This perception was partly due to the fact that some ATWS events in a BWR involve an insertion of positive reactivity. Except for the infrequent occurrence of an unfavorable moderator temperature coefficient, an ATWS event in a PWR is slower, allowing more time for mitigative action.

In spite of this historical perspective for ATWS, it is evident from Figures 8.2 and 8.3 that the ATWS frequencies for the two BWRs are not dramatically higher than for the PWRs. There are several reasons for this. First, plant procedures for dealing with ATWS events have been modified over the past several years, and operator training specifically for these events has improved significantly. Second, the ability to model and analyze ATWS events has improved. More modern neutronic and thermal-hydraulic simulations of the ATWS sequences have calculated lower core power levels during the event than predicted in the past. Further, these calculations indicate that low-pressure injection systems can be used without resulting in significant power oscillations, thus allowing more opportunity for mitigation. Note that for both BWRs and PWRs the frequency of reactor protection system failure remains highly uncertain. Therefore, all comparisons concerning ATWS should be made with caution.

Station blackout accidents contribute a high percentage of the core damage frequency for the BWRs. However, when viewed on an absolute scale, station blackout has a higher frequency at the PWRs than at the BWRs. To some extent this is due to design differences between BWRs and PWRs leading to different susceptibilities. For example, in station blackout accidents, PWRs are potentially vulnerable to reactor coolant pump seal LOCAs following loss of seal cooling, leading to loss of inventory with no method for providing makeup. BWRs, on the other hand, have at least one injection system that does not require ac power. While important, it would be incorrect to imply that the differences noted above are the only considerations that drive the variations in the core damage frequency. Probably more important is the electric power system design at each plant, which is largely independent of the plant type. The station blackout frequency is low at Peach Bottom because of the presence of four diesels that can be shared between units and a maintenance program that led to an order of magnitude reduction in the diesel generator failure rates. Grand Gulf has essentially three trains of emergency ac power for one unit, with one of the trains being both diverse and independent from the other two. These characteristics of the electric power system design tend to dominate any differences in the reactor design. Therefore, a BWR with a below average electric power system reliability could be expected to have a higher station blackout-induced core damage frequency than a PWR with an above average electric power system.

For both BWRs and PWRs, the analyses indicate that, along with electric power, other support systems, such as service water, are quite important. Because these systems vary considerably among plants, caution must be exercised when making statements about generic classes of plants, such as PWRs versus BWRs. Once significant plant-specific vulnerabilities are removed, support-system-driven sequences will probably dominate the core damage frequency of both types of plants. Both types of plants have sufficient redundancy and diversity so as to make multiple independent failures unlikely. Support system failures introduce dependencies among the systems and thus can become dominant.

Boiling Water Reactor Observations

As shown in Figure 8.1, the internal-event core damage frequencies for Peach Bottom and Grand Gulf are extremely low. Therefore, even though dominant plant damage states and contributing failure events can be identified, these items should not be considered as safety problems for the two plants. In fact, these dominating factors should not be overemphasized because, for core damage frequencies below 1E-5, it is possible that other events outside the scope of these internal-event analyses are the ones that actually dominate. In the cases of these two plants, the real perspectives come not from understanding why particular sequences dominate, but rather why all types of sequences considered in the study have low frequencies for these plants.

Previously it was noted that LOCA sequences can be expected to have low frequencies at BWRs because of the numerous systems available to provide coolant injection. While low for both plants, the frequency of LOCAs is higher for Peach Bottom than for Grand Gulf. This is primarily because Grand Gulf is a BWR-6 design with a motor-driven high-pressure core spray system, rather than a steam-driven high-pressure coolant injection system as is Peach Bottom. Motor-driven systems are typically more reliable than steam-driven systems and, more importantly, can operate over the entire range of pressures experienced in a LOCA sequence.

It is evident from Figures 8.2 and 8.4 that station blackout plays a major role in the internal-event core damage frequencies for Peach Bottom and Grand Gulf. Each of these plants has features that tend to reduce the station blackout frequency, some of which would not be present at other BWRs.

Grand Gulf, like all BWR-6 plants, is equipped with an extra diesel generator dedicated to the high-pressure core spray system. While effectively providing a third train of redundant emergency ac power for decay heat removal, the extra diesel also provides diversity, based on a different diesel design and plant location relative to the other two diesels. Because of the aspect of diversity, the analysis neglected common-cause failures affecting all three diesel generators. The net effect is a highly reliable emergency ac power capability. In those unlikely cases where all three diesel generators fail, Grand Gulf relies on a steam-driven coolant injection system that can function until the station batteries are depleted. At Grand Gulf the batteries are sized to last for many hours prior to depletion so that there is a high probability of recovering ac power prior to core damage. In addition, there is a diesel-driven firewater system available that can be used to provide coolant injection in some sequences involving the loss of ac power.

Peach Bottom is an older model BWR that does not have a diverse diesel generator for the high-pressure core spray system. However, other factors contribute to a low station blackout frequency at Peach Bottom. Peach Bottom is a two-unit site, with four diesel generators available. Any one of the four diesels can provide sufficient capacity to power both units in the event of a loss of offsite power, given that appropriate crossties or load swapping between Units 2 and 3 are used. This high level of redundancy is somewhat offset by a less redundant service water system that provides cooling to the diesel generators. Subtleties in the design are such that if a certain combination of diesel generators fails, the service water system will fail, causing the other diesels to fail. In addition, station dc power is needed to start the diesels. (Some emergency diesel generator systems, such as those at Surry, have a separate dedicated dc power system just for starting purposes.) In spite of these factors, the redundancy in the Peach Bottom emergency ac power system is considerable.

While there is redundancy in the ac power system design at Peach Bottom, the most significant factor in the low estimated station blackout frequency relates to the plant-specific data analysis. The plant-specific analysis determined that, because of a high-quality maintenance program, the diesel generators at Peach Bottom had approximately an order of magnitude greater reliability than at an average plant. This factor directly influences the frequency.

Finally, Peach Bottom, like Grand Gulf, has station batteries that are sized to last several hours in the event that the diesel generators do fail. With two steam-driven systems to provide coolant injection and several hours to recover ac power prior to battery depletion, the station blackout frequency is further reduced.

Unlike most PWRs, the response of containment is often a key in determining the core damage frequency for BWRs. For example, at Peach Bottom, there are a number of ways in which containment conditions can affect coolant injection systems. High pressure in containment can lead to closure of primary system relief valves, thus failing low-pressure injection systems, and can also lead to failure of steam-driven high-pressure injection systems due to high turbine exhaust backpressure. High suppression pool temperatures can also lead to the failure of systems that are recirculating water from the suppression pool to the reactor coolant system. If the containment ultimately fails, certain systems can fail because of the loss of net positive suction head in the suppression pool, and also the reactor building is subjected to a harsh steam environment that can lead to failure of equipment located there.

Despite the concerns described in the previous paragraph, the core damage frequency for Peach Bottom is relatively low, compared to the PWRs. There are two major reasons for this. First, Peach Bottom has the ability to vent the wetwell through a 6-inch diameter steel pipe, thus reducing the containment pressure without subjecting the reactor building to steam. While this vent cannot be used to mitigate ATWS and station blackout sequences, it is valuable in reducing the frequency of many other sequences. The second important feature at Peach Bottom is the presence of the control rod drive system, which is not affected by either high pressure in containment or containment failure. Other plants of the BWR-4 design may be more susceptible to containment-related problems if they do not have similar features. For example, some plants have ducting, as opposed to hard piping available for venting. Venting through ductwork may lead to harsh steam environments and equipment failures in the reactor building.*

The Grand Gulf design is generally much less susceptible to containment-related problems than Peach Bottom. The containment design and equipment locations are such that containment rupture will not result in discharge of steam into the building containing the safety systems. Further, the high-pressure core spray system is designed to function with a saturated suppression pool so that it is not affected by containment failure. Finally, there are other systems that can provide coolant injection using water sources other than the suppression pool. Thus, containment failure is relatively benign as far as system operation is concerned, and there is no obvious need for containment venting.

Pressurized Water Reactor Observations

The three PWRs examined in this study reflect much more variety in terms of dominant plant damage states than the BWRs. While the sequence frequencies are generally low for most of the plant damage states, it is useful to understand why the variations among the plants occurred.

For LOCA sequences, the frequency is significantly lower at Surry than at the other two PWRs. A major portion of this difference is directly tied to the additional redundancy available in the injection systems. In addition to the normal high-pressure injection capability, Surry can crosstie to the other unit at the site for an additional source of high-pressure injection. This reduces the core damage frequency due to LOCAs and also certain groups of transients involving stuck-open relief valves.

In addition, at Sequoyah there is a particularly noteworthy emergency core cooling interaction with containment engineered safety features in loss-of-coolant accidents. In this (ice condenser) containment design, the containment sprays are automatically actuated at a very low pressure setpoint, which would be exceeded for virtually all small LOCA events. This spray actuation, if not terminated by the operator can lead to a rapid depletion of the refueling water storage tank at Sequoyah. Thus, an early need to switch to recirculation cooling may occur. Portions of this switchover process are manual at Sequoyah and, because of the timing and possible stressful conditions, leads to a significant human error probability. Thus, LOCA-type sequences are the dominant accident sequence type at Sequoyah.

Station blackout-type sequences have relatively similar frequencies at all three PWRs. Station blackout sequences can have very different characteristics at PWRs than at BWRs. One of the most important findings of the study is the importance of reactor coolant pump seal failures. During station blackout, all cooling to the seals is lost and there is a significant probability that they will ultimately fail, leading to an induced LOCA and loss of inventory. Because PWRs do not have systems capable of providing coolant makeup without ac power, core damage will result if power is not restored. The seal LOCA reduces the time available to restore power and thus increases the station blackout-induced core damage frequency. New seals have been proposed for Westinghouse PWRs and could reduce the core damage frequency if implemented, although they might also increase the likelihood that any resulting accidents would occur at high pressure, which has implications for the accident progression analysis. (See Section C.14 of Appendix C for a more detailed discussion of reactor coolant seal performance.)

Apart from the generic reactor coolant pump seal question, station blackout frequencies at PWRs are determined by the plant-specific electric power system design and the design of other support systems. Battery depletion times for the three PWRs were projected to be shorter than for the two BWRs. A particular characteristic of the

*The staff is presently undertaking regulatory action to require hard pipe vents in all BWR Mark I plants.

Surry plant is a gravity-fed service water system with a canal that may drain during station blackout, thus failing containment heat removal. When power is restored, the canal must be refilled before containment heat removal can be restored.

The dominant accident sequence type at Zion is not a station blackout, but it has many similar characteristics. Component cooling water is needed for operation of the charging pumps and high-pressure safety injection pumps at Zion. Loss of component cooling water (or loss of service water, which will also render component cooling water inoperable) will result in loss of these high-pressure systems. This in turn leads to a loss of reactor coolant pump seal injection. Simultaneously, loss of component cooling water will also result in loss of cooling to the thermal barrier heat exchangers for the reactor coolant pump seals. Thus, the reactor coolant pump seals will lose both forms of cooling. As with station blackout, loss of component cooling water or service water can both cause a small LOCA (by seal failure) and disable the systems needed to mitigate it. The importance of this scenario is increased further by the fact that the component cooling water system at Zion, although it uses redundant pumps and valves, delivers its flow through a common header. The licensee for the Zion plant has made procedural changes and is also considering both the use of new seal materials and the installation of modifications to the cooling water systems. These measures, which are discussed in more detail in Chapter 7, reduce the importance of this contributor.

ATWS frequencies are generally low at all three of the PWRs. This is due to the assessed reliability of the shutdown systems and the likelihood that only slow-acting, low-power-level events will result.

While of low frequency, it is worth noting that interfacing-system LOCA (V) and steam generator tube rupture (SGTR) events do contribute significantly to risk for the PWRs. This is because they involve a direct path for fission products to bypass containment. There are large uncertainties in the analyses of these two accident types, but these events can be important to risk even at frequencies that may be one or two orders of magnitude lower than other sequence types.

During the past few years, most Westinghouse PWRs have developed procedures for using feed and bleed cooling and secondary system blowdown to cope with loss of all feedwater. These procedures have led to substantial reductions in the frequencies of transient sequences involving the loss of main and auxiliary feedwater. Appropriate credit for these actions was given in these analyses. However, there are plant-specific features that will affect the success rate of such actions. For example, the loss of certain power sources (possibly only one bus) or other support systems can fail power-operated relief valves (PORVs) or atmospheric dump valves or their block valves at some plants, precluding the use of feed and bleed or secondary system blowdown. Plants with PORVs that tend to leak may operate for significant periods of time with the block valves closed, thus making feed and bleed less reliable. On the other hand, if certain power failures are such that open block valves cannot be closed, then they cannot be used to mitigate stuck-open PORVs. Thus, both the system design and plant operating practices can be important to the reliability assessment of actions such as feed and bleed cooling.

8.4.4 External Events

The frequency of core damage initiated by external events has been analyzed for two of the plants in this study, Surry and Peach Bottom (Ref. 8.1 (Part 3) and Ref. 8.2 (Part 3)). The analysis examined a broad range of external events, e.g., lightning, aircraft impact, tornados, and volcanic activity (Ref. 8.8). Most of these events were assessed to be insignificant contributors by means of bounding analyses. However, seismic events and fires were found to be potentially major contributors and thus were analyzed in detail.

Figures 8.7 and 8.8 show the results of the core damage frequency analysis for seismic- and fire-initiated accidents, as well as internally initiated accidents, for Surry and Peach Bottom, respectively. Examination of these figures shows that the core damage frequency distributions of the external events are comparable to those of the internal events. It is evident that the external events are significant in the total safety profile of these plants.

Seismic Analysis Observations

The analysis of the seismically induced core damage frequency begins with the estimation of the seismic hazard, that is, the likelihood of exceeding different earthquake ground-motion levels at the plant site. This is a difficult, highly judgmental issue, with little data to provide verification of the various proposed geologic and seismologic models.

The sciences of geology and seismology have not yet produced a model or group of models upon which all experts agree. This study did not itself produce seismic hazard curves, but instead made use of seismic hazard curves for Peach Bottom and Surry that were part of an NRC-funded Lawrence Livermore National Laboratory project that resulted in seismic hazard curves for all nuclear power plant sites east of the Rocky Mountains (Ref. 8.9).

In addition, the Electric Power Research Institute (EPRI) developed a separate set of models (Ref. 8.10). For purposes of completeness and comparison, the seismically induced core damage frequencies were also calculated based upon the EPRI methods. Both sets of results, which are presented in Figures 8.5 through 8.8, were used in this study. More detailed discussion of methods used in the seismic analysis is provided in Appendix A; Section C.11 of Appendix C provides more detailed perspectives on the seismic issue as well.

As can be seen in Figures 8.5 and 8.6, the shapes of the seismically induced core damage probability distributions are considerably different from those of the internally initiated and fire-initiated events. In particular, the 5th to 95th percentile range is much larger for the seismic events. In addition, as can be seen in Figures 8.7 and 8.8, the wide disparity between the mean and the median and the location of the mean relatively high in the distribution indicate a wide distribution with a tail at the high end but peaked much lower down. (This is a result of the uncertainty in the seismic hazard curve.)

It can be clearly seen that the difference between the mean and median is an important distinction. The mean is the parameter quoted most often, but the bulk of the distribution is well below the mean. Thus, although the mean is the "center of gravity" of the distribution (when viewed on a linear rather than logarithmic scale), it is not very representative of the distribution as a whole. Instead, it is the lower values that are more probable. The higher values are estimated to have low probability, but, because of their great distance from the bulk of the distribution, the mean is "pulled up" to a relatively high value. In a case such as this, it is particularly evident that the entire distribution, not just a single parameter such as the mean or the median, must be considered when discussing the results of the analysis.

1. Surry Seismic Analysis

The core damage frequency probability distributions, as calculated using the Livermore and EPRI methods, have a large degree of overlap, and the differences between the means and medians of the two resulting distributions are not very meaningful because of the large widths of the two distributions.

The breakdown of the Surry seismic analysis into principal contributors is reasonably similar to the results of other seismic PRAs for other PWRs. The total core damage frequency is dominated by loss of offsite power transients resulting from seismically induced failures of the ceramic insulators in the switchyard. This dominant contribution of ceramic insulator failures has been found in virtually all seismic PRAs to date.

A site-specific but significant contributor to the core damage frequency at Surry is failure of the anchorage welds of the 4 kV buses. These buses play a vital role in providing emergency ac electrical power since offsite power as well as emergency onsite power passes through these buses. Although these welded anchorages have more than adequate capacity at the safe shutdown earthquake (SSE) level, they do not have sufficient margin to withstand (with high reliability) earthquakes in the range of four times the SSE, which are contributing to the overall seismic core damage frequency results.

Similarly, a substantial contribution is associated with failures of the diesel generators and associated load center anchorage failures. These anchorages also may not have sufficient capacity to withstand earthquakes at levels of four times the SSE.

Another area of generic interest is the contribution due to vertical flat-bottomed storage tanks, e.g., refueling water storage tanks and condensate storage tanks. Because of the nature of their configuration and field erection practices, such tanks have often been calculated to have relatively smaller margin over the SSE than most components in commercial nuclear power plants. Given that all PWRs in the United States use the refueling water storage tank as the primary source of emergency injection water (and usually the sole source until the recirculation phase of ECCS begins), failure of the refueling water storage tank can be expected to be a substantial contributor to the seismically induced core damage frequency.

2. Peach Bottom Seismic Analysis

As can be seen in Figure 8.9, the dominant contributor in the seismic core damage frequency analysis is a transient sequence brought about by loss of offsite power. The loss of offsite power is due to seismically induced failures of onsite ac power. Peach Bottom has four emergency diesel generators, all shared between the two units, and four station batteries per unit. Thus, there is a high degree of redundancy. However, all diesels require cooling provided by the emergency service water system, and failure to provide this cooling will result in failure of all four diesels.

There is a variety of seismically induced equipment failures that can fail the emergency service water system and result in a station blackout. These include failure of the emergency cooling tower, failures of the 4 kV buses (in the same manner as was found at Surry), and failures of the emergency service water pumps or the emergency diesel generators themselves. The various combinations of these failures result in a large number of potential failure modes and give rise to a relatively high frequency of core damage based on station blackout. None of these equipment failure probabilities is substantially greater than would be implied by the generic fragility data available. However, the high probability of exceedance of larger earthquakes (as prescribed by the hazard curves for this site) results in significant contributions of these components to the seismic risk.

Fire Analysis Observations

The core damage likelihood due to a fire in any particular area of the plant depends upon the frequency of ignition of a fire in the area, the amount and nature of combustible material in that area, the nature and efficacy of the fire-suppression systems in that area, and the importance of the equipment located in that area, as expressed in the potential of the loss of that equipment to cause a core damage accident sequence. The methods used in the fire analysis are described in Appendix A and in Reference 8.7; Section C.12 of Appendix C provides additional perspectives on the fire analysis.

1. Surry Fire Analysis

Figure 8.10 shows the dominant contributors to core damage frequency resulting from the Surry fire analysis. The dominant contributor is a transient resulting in a reactor coolant pump seal LOCA, which can lead to core damage. The scenario consists of a fire in the emergency switchgear room that damages power or control cables for the high-pressure injection and component cooling water pumps. No additional random failures are required for this scenario to lead to core damage. It should be noted that credit was given for existing fire-suppression systems and for recovery by crossconnecting high-pressure injection from the other unit. The importance of this scenario is evident in Figure 8.11, which breaks down the fire-induced core damage frequency by location in the plant. The most significant physical location is the emergency switchgear room. In this room, cable trays for the two redundant power trains were run one on top of the other with approximately 8 inches of vertical separation in a number of plant areas, which gives rise to the common vulnerability of these two systems due to fire. In addition, the Halon fire-suppression system in this room is manually actuated.

The other principal contributor is a spuriously actuated pressurizer PORV. In this scenario, fire-related component damage in the control room includes control power for a number of safety systems. Full credit was given for independence of the remote shutdown panel from the control room except in the case of PORV block valves; discussions with utility personnel indicated that control power for these valves was not independently routed.

2. Peach Bottom Fire Analysis

Figure 8.10 shows the mechanisms by which fire leads to core damage in the Peach Bottom analysis. Station blackout accidents are the dominant contributor, with substantial contributions also coming from fire-induced transients and losses of offsite power. The relative importance of the various physical locations is shown in Figure 8.12.

It is evident from Figure 8.12 that control room fires are of considerable significance in the fire analysis of this plant. Fires in the control room were divided into two scenarios, one for fires initiating in the reactor core isolation cooling (RCIC) system cabinet and one for all others. Credit was given for automatic cycling of the RCIC system unless the fire initiated within its control panel. Because of the cabinet configuration within the control room, the fire was assumed not to spread and damage any components outside the cabinet where the fire initiated. The analysis gave credit for the possibility of quick extinguishing of the fire within the applicable cabinet since the control room is continuously occupied. However, should these efforts fail, even with high ventilation rates, these scenarios postulate forced abandonment of the control room due to smoke from the fire and subsequent plant control from the remote shutdown panel.

The cable spreading room below the control room is significant but not dominant in the fire analysis. The scenario of interest is a fire-induced transient coupled with fire-related failures of the control power for the high-pressure coolant injection system, the reactor core isolation cooling system, the automatic depressurization system, and the control rod drive hydraulic system. The analysis gave credit to the automatic CO_2 fire-suppression system in this area.

The remaining physical areas of significance are the emergency switchgear rooms. The fire-induced core damage frequency is dominated by fire damage to the emergency service water system in conjunction with random failures coupled with fire-induced loss of offsite power. In all eight emergency switchgear rooms (four shared between the two units), both trains of offsite power are routed. It was noted that in each of these areas there are breaker cubicles for the 4 kV switchgear with a penetration at the top that has many small cables routed through it. These penetrations were inadequately sealed, which would allow a fire to spread to cabling that was directly above the switchgear room. This cabling was a sufficient fuel source for the fire to cause a rapid formation of a hot gas layer that would then lead to a loss of offsite power. Since both offsite power and the emergency service water systems are lost, a station blackout would occur.

Perspectives: General Observations on Fire Analysis

Figures 8.7 and 8.8 clearly indicate that fire-initiated core damage sequences are significant in the total probabilistic analysis of the two plants analyzed. Moreover, these analyses already include credit for the fire protection programs required by Appendix R to 10 CFR Part 50.

Although the two plants are of completely different design, with completely different fire-initiated core damage scenarios, the possibility of fires in the emergency switchgear areas is important in both plants. The importance of the emergency switchgear room at Surry is particularly high because of the seal LOCA scenario. Further, the importance of the control room at Surry is comparable to that of the control room at Peach Bottom.

This is not surprising in view of the potential for simultaneous failure of several systems by fires in these areas. Thus, in the past such areas have generally received particular attention in fire protection programs. It should also be noted that the significance of various areas also depends upon the scenario that leads to core damage. For example, the importance of the emergency switchgear room at Surry could be altered (if desired) not only by more fire protection programs but also by changes in the probability of the reactor coolant pump seal failure.

REFERENCES FOR CHAPTER 8

8.1 R. C. Bertucio and J. A. Julius, "Analysis of Core Damage Frequency: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 3, Revision 1, SAND86-2084, April 1990.

8.2 A. M. Kolaczkowski et al., "Analysis of Core Damage Frequency: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4550, Vol. 4, Revision 1, SAND86-2084, August 1989.

8.3 R. C. Bertucio and S. R. Brown, "Analysis of Core Damage Frequency: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 5, Revision 1, SAND86-2084, April 1990.

8.4 M. T. Drouin et al., "Analysis of Core Damage Frequency: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 6, Revision 1, SAND86-2084, September 1989.

8.5 M. B. Sattison and K. W. Hall, "Analysis of Core Damage Frequency: Zion Unit 1," Idaho National Engineering Laboratory, NUREG/CR-4550, Vol. 7, Revision 1, EGG-2554, May 1990.

8.6 USNRC, "Reactor Safety Study—An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

8.7 H. J. C. Kouts et al., "Special Committee Review of the Nuclear Regulatory Commission's Severe Accident Risks Report (NUREG-1150)," NUREG-1420, August 1990.

8.8 M. P. Bohn and J. A. Lambright, "Procedures for the External Event Core Damage Frequency Analyses for NUREG-1150," Sandia National Laboratories, NUREG/CR-4840, SAND88-3102, November 1990.

8.9 D. L. Bernreuter et al., "Seismic Hazard Characterization of 69 Nuclear Power Sites East of the Rocky Mountains," Lawrence Livermore National Laboratory, NUREG/CR-5250, Vols. 1-8, UCID-21517, January 1989.

8.10 Seismicity Owners Group and Electric Power Research Institute, "Seismic Hazard Methodology for the Central and Eastern United States," Electric Power Research Institute, EPRI NP-4726, July 1986.

9. PERSPECTIVES ON ACCIDENT PROGRESSION AND CONTAINMENT PERFORMANCE

9.1 Introduction

The consequences of severe reactor accidents depend greatly on containment safety features and containment performance in retaining radioactive material. The early failure of the containment structures at the Chernobyl power plant contributed to the size of the environmental release of radioactive material in that accident. In contrast, the radiological consequences of the Three Mile Island Unit 2 (TMI-2) accident were minor because overall containment integrity was maintained and bypass was small. Normally three barriers (the fuel rod cladding, the reactor coolant system pressure boundary, and the containment pressure boundary) protect the public from the release of radioactive material generated in nuclear fuel. In most core meltdown scenarios, the first two barriers would be progressively breached, and the containment boundary represents the final barrier to release of radioactivity to the environment. Maintaining the integrity of the containment can affect the source term by orders of magnitude. The NRC's 1986 reassessment of source term issues reaffirmed that containment performance "is a major factor affecting source terms" (Ref. 9.1).

In most severe accident sequences, the ability of a containment boundary to maintain integrity is determined by two factors: (1) the magnitude of the loads, and (2) the response to those loads of the containment structure and the penetrations through the containment boundary. Although there is no universally accepted definition of containment failure, it does not necessarily imply gross structural failure. For risk purposes, containment is considered to have failed to perform its function when the leak rate of radionuclides to the environment is substantial. Thus, failure could occur as the result of a structural failure of the containment, tearing of the containment liner, or a high rate of a leakage through a penetration. Finally, valves that are open during normal operation may not close properly when the accident occurs. Failure of the containment isolation system can result in leakage of radioactive material to a secondary building or directly to the environment.

In some accidents, the containment building is completely bypassed. In interfacing-system loss-of-coolant accidents (LOCAs), check valves isolating low-pressure piping fail, and the piping connected to the reactor coolant system fails outside the containment. The radionuclides can escape to secondary buildings through the reactor coolant system piping without passing through the containment. A similar bypass can occur in a core meltdown sequence initiated by the rupture of a steam generator tube in which release is through relief valves on the steam line from the failed steam generators.

Although the five plants analyzed in the present study were selected to span the basic types of containment design used in the United States, it cannot be assumed that the containment performance results obtained are characteristic of a class of plants. The loads in an accident sequence, the relative frequencies of specific accident sequences, and the load level at which the containment fails can all be influenced by design details that vary among reactors within a class of containments. (Additional discussion of the extrapolability of PRA results is provided in Chapter 13.)

9.2 Summary of Results

If the containment function is maintained in a severe accident, the radiological consequences will be minor. If the containment function does fail, the timing of failure can be very important. The longer the containment remains intact relative to the time of core melting and radionuclide release from the reactor coolant system, the more time is available to remove radioactive material from the containment atmosphere by engineered safety features or natural deposition processes. Delay in containment failure or containment bypass also provides time for protective action, a very important consideration in the assessment of possible early health effects. Thus, in evaluating the performance of a containment, it is convenient to consider no failure, late failure, bypass, and early failure of containment as separate categories characterizing different degrees of severity. For those plants in which intentional venting is an option, this is also represented as a separate category.

Not all accident sequences that involve core damage would necessarily progress to vessel failure, as illustrated by the TMI-2 accident. The operator may recover a critical system (such as by the return of offsite power) or the state of the plant may change (for example, the system pressure may fall to a point where low-pressure emergency coolant systems can be activated) allowing the core to be recovered and the accident to be terminated. The likelihood of containment failure in terminated accidents is typically less than in accidents involving vessel failure, and the radiological consequences are usually very small.

9.2.1 Internal Events

The probability of early containment failure and vessel breach conditional on the indicated class of sequence (and the mean frequency of the class) is illustrated in Figure 9.1 for three classes of accident sequences in the pressurized water reactors (PWRs) analyzed in this study and in Figure 9.2 for three classes of accident sequences in the boiling water reactors (BWRs) analyzed (Refs. 9.2 through 9.6). Containment bypass scenarios are not included in these figures, and the results are for internally initiated accidents. For different plant designs, the nature of the loads and the response of the containment are different, even for the same accident class.

The predicted likelihoods of early containment failure in the Zion (large, dry design) plant and the Surry (subatmospheric design) plant are quite small (mean value of about 1 percent). The principal mechanisms leading to these failures are loads resulting from high-pressure melt ejection in accident sequences with high reactor coolant system (RCS) pressures (at time of vessel breach) and in-vessel steam explosions in sequences with low RCS pressures at vessel breach. Both phenomena involve substantial uncertainties.

The principal reason that the probability of early containment failure from loads at vessel breach is so small in the Surry and Zion analyses is that the reactor coolant system is not likely to be at high pressure when vessel meltthrough occurs. Some of the mechanisms that were found to be effective in depressurizing the vessel are hot leg or surge line failure at elevated temperature, failure of a reactor coolant pump seal, or a stuck-open relief valve. If an extreme case at Surry is selected, which is a large core fraction ejected, a dry cavity, no sprays, a large hole in the vessel, and high reactor coolant system pressure, the conditional probability of containment failure is approximately 30 percent. However, this is a very unlikely case. For cases with small holes in the reactor vessel and a small or intermediate fraction of the core ejected, which are much more likely, the probability of containment failure is a few percent or less.

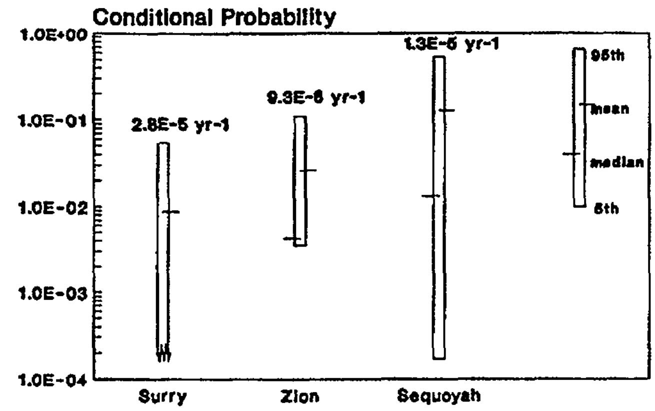
For accident sequences at Surry and Zion in which core uncovery is initiated with the reactor coolant system at high pressure, the probability of overheating and rupturing steam generator tubes after the onset of core damage, with subsequent bypass of the containment, is of the same magnitude as the probability of early containment failure from high-pressure ejection of core debris with direct containment heating. In Figure 9.1, the smaller spread in uncertainty in the downward direction for the Zion plant is due to the higher frequency of containment isolation failure, which establishes a lower bound for the distribution.

The results for the Sequoyah plant indicate that early containment failure is somewhat more likely for ice condenser designs than for large, high-pressure containments. The mean likelihood of early failure is approximately 12 percent (8 percent includes vessel breach, 4 percent does not). Early containment failure is primarily the result of loads at vessel failure. For scenarios in which the vessel is at high pressure at the time of vessel breach, early failure results from overpressurization (including the pressure load from hydrogen burning) or from direct attack of the containment by hot debris following failure of the seal table. If the vessel is at low pressure at vessel breach, the principal failure mechanism is overpressurization.

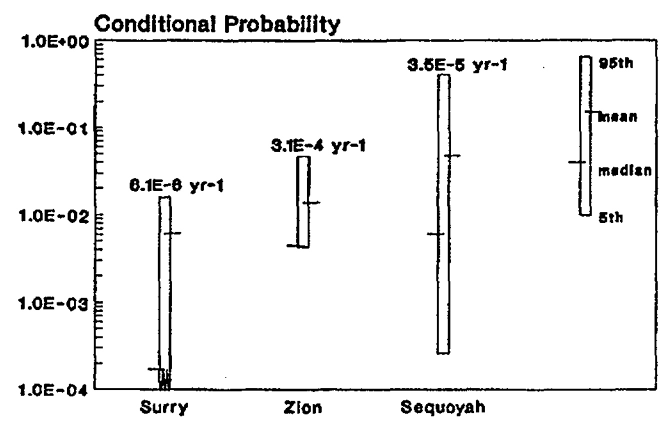
The predicted probability of early failure of the Peach Bottom and Grand Gulf pressure-suppression containments is substantially higher than for the PWR containment designs. For Grand Gulf, the mean probability of early failure is approximately 50 percent while at Peach Bottom the mean probability of early failure is about 56 percent.

In the Peach Bottom (Mark I design) plant, failure is predicted to occur primarily in the drywell as a result of direct attack by molten core debris. Drywell rupture due to pedestal failure or rapid overpressurization (more quickly than the water columns in the vent lines can be cleared) is also an important contributor to early containment failure. If failure occurs in the drywell, releases of radionuclides from fuel after vessel failure will not pass through the suppression pool. Late failure of containment is also most likely to occur in the drywell but in the form of prolonged leakage past the drywell head.

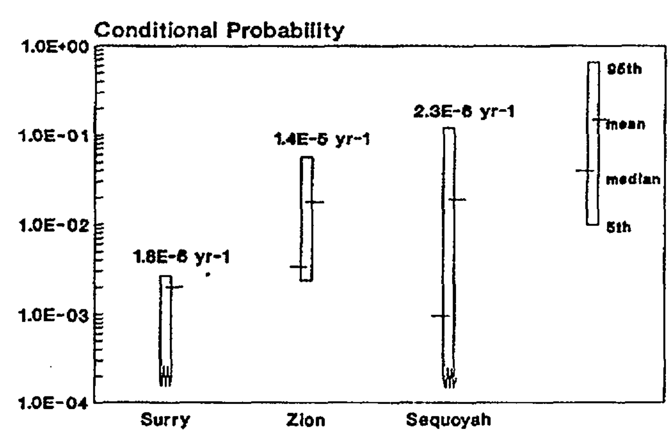
At Grand Gulf, early containment failure in station blackout is dominated by hydrogen deflagrations. Hydrogen detonations are also small contributors to early failure. For short-term station blackouts (the dominant plant damage state groups), the conditional probability of early



a. Station blackout

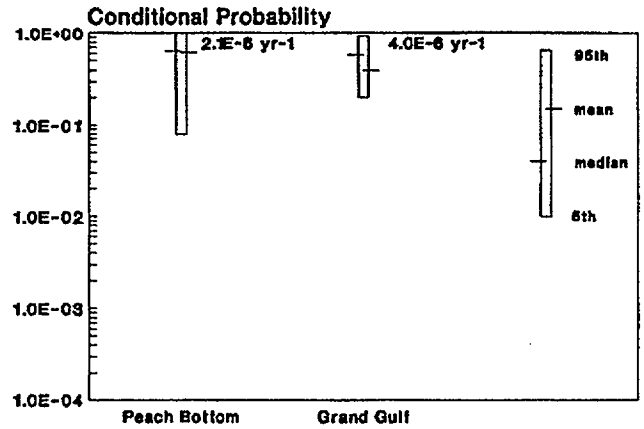


b. Loss-of-coolant accidents

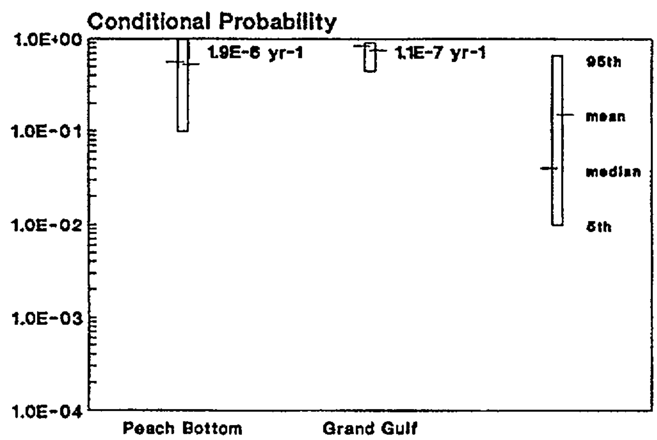


c. Transients

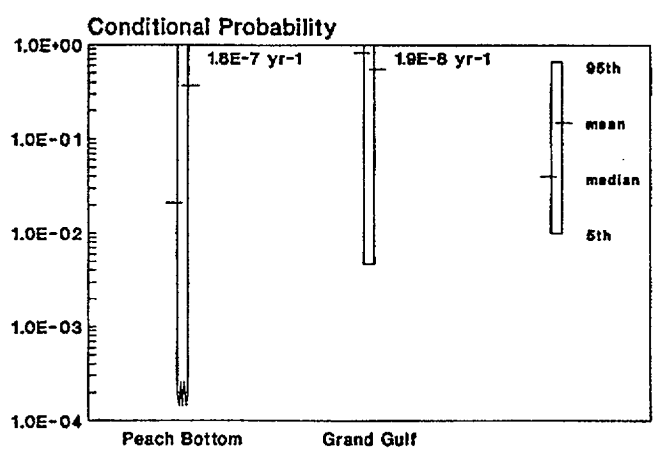
Figure 9.1 Conditional probability of early containment failure for key plant damage states (PWRs).



a. Station blackout



b. Anticipated transients without scram



c. Transients

Figure 9.2 Conditional probability of early containment failure for key plant damage states (BWRs).

containment failure is 50 percent. About half of the early containment failures occur before vessel breach, and the other half occur at or shortly after vessel breach. For the long-term station blackouts, the mean conditional probability of early containment failure is 85 percent.

The probability of drywell failure at Grand Gulf is somewhat less than that of containment failure and occurs in approximately one-half the early containment failures. Drywell failures before vessel breach result from rapid hydrogen deflagrations in the wetwell. At the time of vessel breach, however, drywell failures are primarily from drywell pressurization loads at vessel breach (steam blowdown, direct containment heating, ex-vessel steam explosions, and hydrogen combustion). Failure of the drywell is more likely when vessel breach occurs with the vessel at high pressure.

Intentional venting of the containment was considered to prevent overpressurization failure of the containment for both Peach Bottom and Grand Gulf. The mean probability of sequences in which containment venting occurs and no containment failure occurs is approximately 10 percent for Peach Bottom station blackout sequences and 4 percent for Grand Gulf. The values are small, mostly because of the high probability of early failure mechanisms for which venting is ineffective. Furthermore, for the short-term station blackout plant damage state that dominates the core melt frequency at Grand Gulf, ac power is not available initially to permit venting.

Figure 9.3 illustrates the frequency of early failure or bypass of containment (the two types of failure with the potential for a large release of radionuclides) for internally initiated accidents in each of the five plants. (Peach Bottom scenarios in which the containment has been vented but subsequent early containment failure has occurred are categorized as early containment failures.) Note that, on a basis of absolute frequency, early containment failure or bypass for the BWR designs analyzed is similar to that of the PWRs because of the lower predicted frequency of core damage in the BWRs.

The relative probabilities of early containment failure, bypass, late failure, venting, and no containment failure are illustrated in Figure 9.4 for each of the plants. For the Surry plant, the likelihood of bypass, an interfacing-system LOCA, or steam generator tube rupture is somewhat greater than that of early failure from severe accident loads. In Figure 9.4, the capability of the Zion plant to avoid a large early release of radioactive material appears to be particularly good because of the small fraction of failures that result in either early failure or bypass.

It should be noted that the averaging of containment failure mode probabilities for different plant damage states can be misleading. To a large degree, the relative probability of bypass at Zion is substantially smaller than at Surry because the frequency of plant damage states, other than the interfacing-system LOCA, is higher. On an absolute frequency scale, as shown in Figure 9.3, the performances of the Surry and Zion containments in severe accidents are quite similar. In Sequoyah, the probability of early failure is somewhat larger than for the other PWRs analyzed and on a frequency-weighted mean basis is essentially the same as for bypass. The most likely outcome for these plants is that the containment will not fail.

Using early containment failure or containment bypass as a measure for comparison, the performance of the two BWR containments analyzed does not appear as good as the performance of the PWR containments. It is important to recognize that early containment failure or bypass is a prerequisite for a large release of radionuclides, but that mitigative features within the plant can substantially limit the release that occurs. This is particularly true for the pressure-suppression containment designs, where the suppression pool or ice condenser can retain radionuclides even if the containment has failed. (The BWR frequency of bypass is assessed to be quite small. Therefore, only early failures (with the potential for some radionuclide scrubbing by the suppression pool) are important.) The frequency of release of different quantities of radionuclides is discussed in Chapter 10.

9.2.2 External Events

Plant damage states that result from external events are quite similar to those that arise from internally initiated accidents except that their relative frequencies differ substantially. In addition, containment status may be affected by the initiating event. Figure 9.5 illustrates the relative probabilities of early containment failure, bypass, late failure, venting, and no failure (no vessel breach or vessel breach with no containment failure) for the two plants for which external-event analyses were performed. The results for internal initiators, fire, and seismic are compared in the figure. The importance of early containment failure relative to the importance of bypass is reversed in the Surry

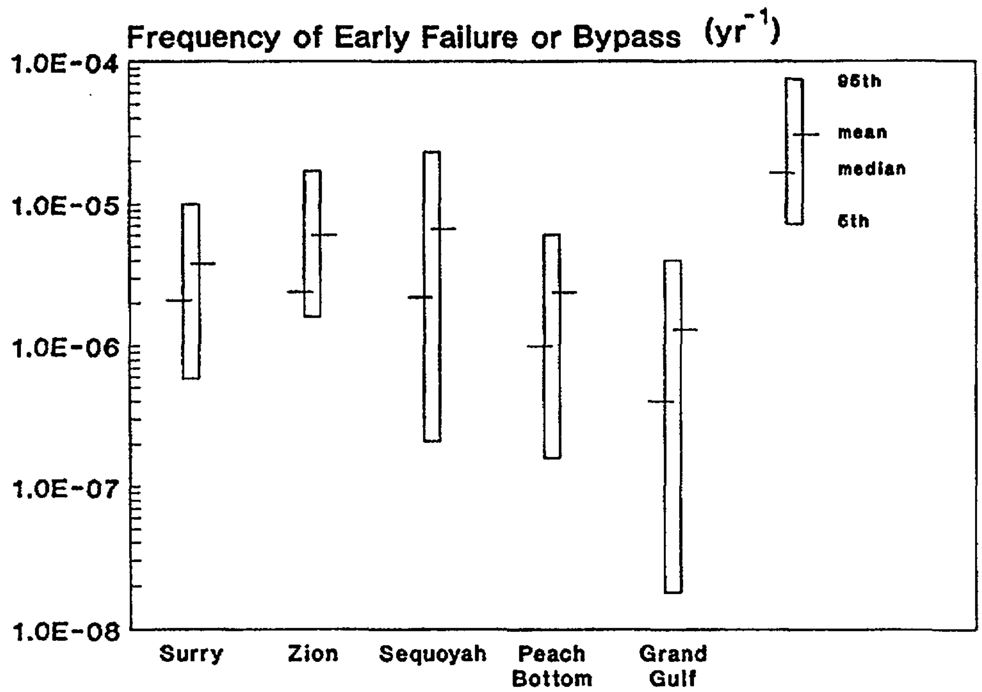


Figure 9.3 Frequency of early containment failure or bypass (all plants).

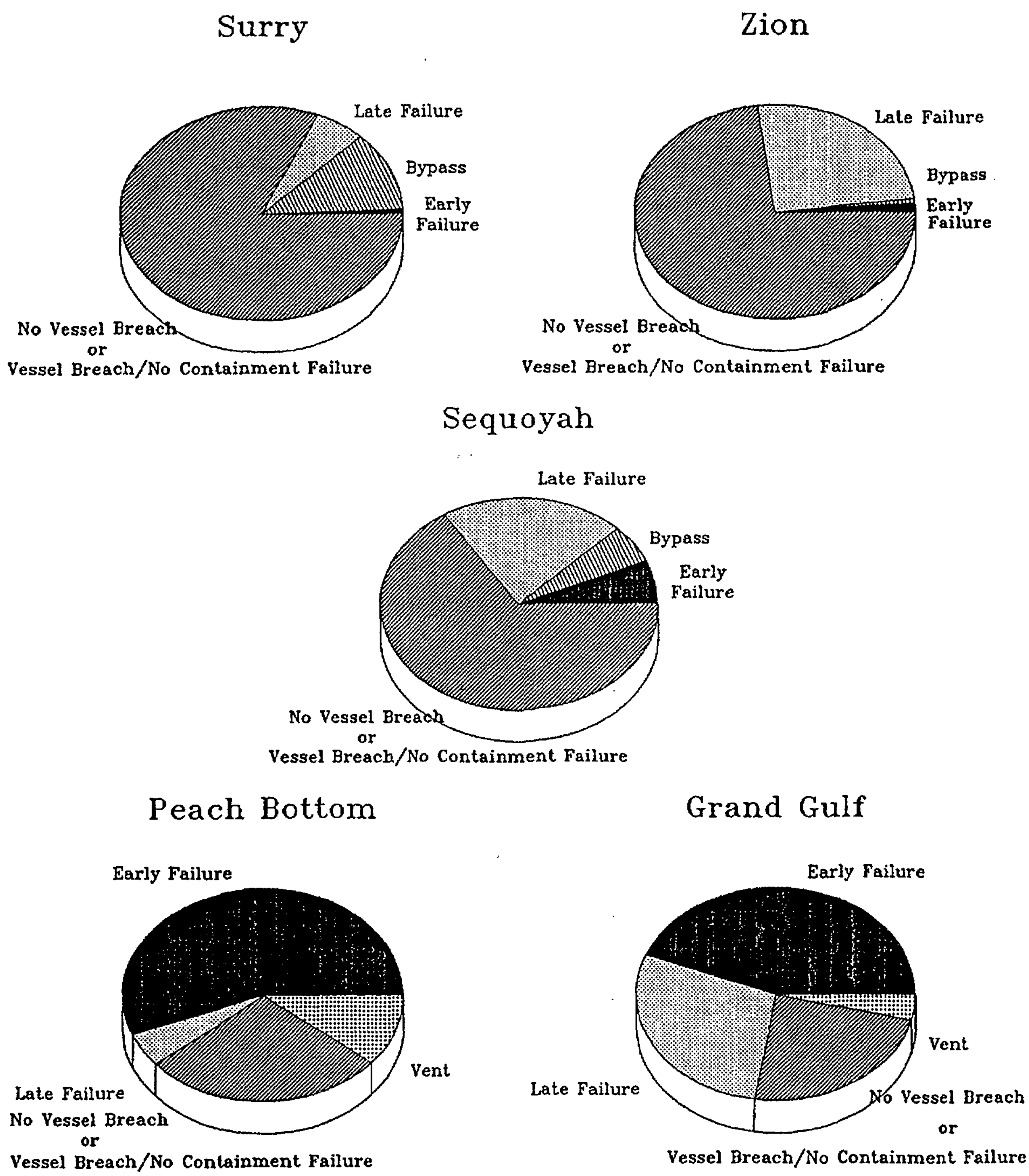


Figure 9.4 Relative probability of containment failure modes (internal events).

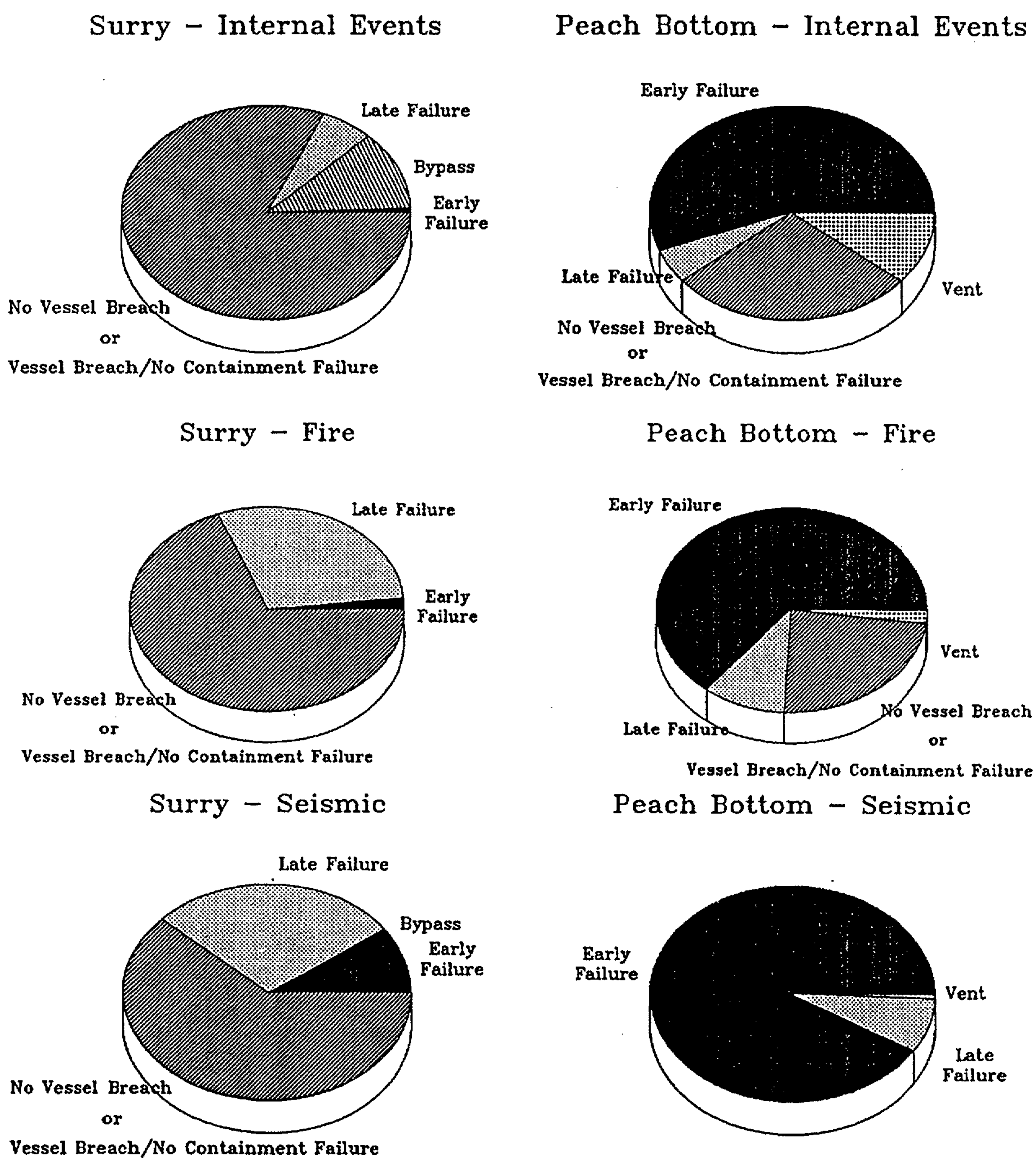


Figure 9.5 Relative probability of containment failure modes (internal and external events, Surry and Peach Bottom).

external-event analysis compared to the internal analysis. In the seismic analysis, the conditional probability of early failure is predicted to increase significantly (to approximately 8 percent). The increased failure likelihood is associated with substantial motion of the reactor coolant system components in an earthquake and resulting damage to the containment. In the fire analysis, there are no externally initiated bypass accidents, the likelihood of bypass induced by overheating of steam generator tubes is assessed to be negligible, and there is only a very slight increase in early containment failure.

Perspectives on the differences between external-event and internal-event containment performance for the Peach Bottom plant are similar to those described for Surry. In the fire analysis, some increase in early containment failure is predicted. In the fire sequences, there is a reduced potential for the recovery of ac power, which results in a reduced probability of injection recovery and an increased likelihood of drywell shell meltthrough.

In the BWR seismic analysis, the probability of containment survival in a severe accident is small; the increased likelihood of early containment failure is the result of substantial motion of the reactor vessel and subsequent damage to the containment during a major earthquake (well beyond the plant's design level) and a reduced recovery potential that increases the likelihood of containment failure as described for the fire sequences.

9.2.3 Additional Summary Results

Based on the results of the five-plant risk analyses summarized in Chapters 3 through 7, and discussed in detail in References 9.2 through 9.6, the following perspectives on containment performance in severe accidents can be drawn.

Zion and Surry Plants (Large, Dry and Subatmospheric Designs)

- Large, dry and subatmospheric containment designs appear to be quite robust in their ability to contain severe accident loads. This study shows a high likelihood of maintaining integrity throughout the early phases of severe accidents in which the potential for a large release of radionuclides is greatest. The uncertainties in describing the magnitude of severe accident loads at vessel breach for pressurized scenarios and the likelihood of depressurization prior to lower head failure are large, however.

- Containment bypass sequences (severe accidents initiated by steam generator tube ruptures, tube ruptures induced by hot circulating gases, or interfacing-system LOCAs) represent a substantial fraction of high-consequence accidents. The absolute frequency of these types of failure is small, however.

- The potential exists for the arrest of core degradation in a significant fraction of core damage scenarios within the reactor vessel as the result of recovery procedures (such as in the TMI-2 accident). The likelihood of containment failure is very small in these scenarios.

- A substantial likelihood exists that the containment will remain intact even if the accident progresses beyond the point of lower head failure.

- The likelihood of early containment failure in seismic events is higher than for internally initiated accidents.

Sequoyah Plant (Ice Condenser Design)

- The likelihood of early failure in a severe accident for the Sequoyah plant is higher than for the large, dry and subatmospheric designs but is less than for the BWRs analyzed. Early failure is primarily associated with loads imposed at the time of vessel breach (from a number of mechanisms, including direct containment heating and hydrogen combustion).

- Containment rupture from high overpressure loads at the time of vessel breach is likely to result in significant damage to the containment wall and effective bypass of the ice bed.

- Containment bypass is potentially an important contributor to the frequency of a large early release of radioactive material.

- The high likelihood of a deeply flooded reactor cavity plays an important role in mitigating severe accident consequences at Sequoyah. The deeply flooded cavity assists in reducing the loads at vessel breach, in preventing direct attack of molten fuel debris on the containment wall, and in avoiding molten core-concrete interactions.

- There is substantial potential for the arrest of core damage prior to vessel failure. There is, however, some likelihood of containment failure from hydrogen combustion events.

- A substantial likelihood exists for containment integrity to be preserved throughout a severe accident, even if the accident progresses beyond vessel breach.

Peach Bottom Plant (Mark I Design)

- The analyses indicate a substantial likelihood for early drywell failure in severe accident scenarios, primarily as the result of direct attack of the drywell shell by molten core debris.

- Considerable uncertainty exists regarding the likelihood of failure of the drywell as the result of direct attack by core debris. Although this is the dominant failure mechanism in the analyses, other loads on the drywell can lead to early drywell failure, such as rapid overpressurization of the drywell. A sensitivity study was performed in which the drywell meltthrough mechanism of failure was eliminated. The resulting reduction in mean early containment failure probability was from 0.56 to 0.2 (Ref. 9.3).

- The principal benefit of wetwell venting indicated by the study is in the reduction of the core damage frequency. Although venting is not effective in eliminating some early drywell failure mechanisms, venting could eliminate other sequences that would result in overpressure failure of the containment.

- There is substantial potential for the arrest of core damage prior to vessel failure. The likelihood of containment failure in arrested scenarios is small.

- The likelihood of early containment failure is higher for fire and seismic events than internally initiated accidents because of the decreased likelihood of ac and dc recovery resulting in higher drywell shell meltthrough probabilities.

Grand Gulf Plant (Mark III Design)

- Grand Gulf containment was predicted to fail at or before vessel breach in a substantial fraction of severe accident sequences. Hydrogen deflagration is the principal mechanism for early containment failure.

- Failure of the integrity of the drywell is predicted to accompany containment failure in approximately one-half the sequences involving early containment failure (resulting in bypass of the suppression pool for radionuclides released after vessel breach). Drywell failure is primarily the result of loads from rapid combustion events prior to reactor vessel breach and loads at vessel breach associated with overpressurization by direct containment heating, ex-vessel steam explosions, and hydrogen combustion in the wetwell region. Scrubbing of releases occurring before vessel breach can still occur in sequences in which the drywell fails and the suppression pool is eventually bypassed.

- There is a large potential for the arrest of core damage prior to vessel failure. If large quantities of hydrogen are produced in the process of recovery, hydrogen combustion could result in containment failure.

- Venting was not found to be particularly effective in preventing containment failure for accident scenarios involving core damage. Furthermore, venting was not as effective in reducing core damage frequency in Grand Gulf as it was in Peach Bottom.

9.3 Comparison with Reactor Safety Study

Prior to the time the Reactor Safety Study (RSS) (Ref. 9.7) analyses were undertaken, there had been no relevant experimentation or modeling of either the loads produced in a severe accident or the response of a containment to loads exceeding the design basis. As a result, the characterization of containment performance in the RSS is simplistic in comparison to the present study.

Containment Failure Modes

Figure 9.6 compares estimates for the present study with those of the RSS for the cumulative failure probability as a function of internal pressure for the Surry plant. The current study indicates that the Surry containment is substantially stronger than did the RSS characterization. In the RSS analyses, failure was assumed to involve rupture of the containment with substantial leakage to the environment. The current study subdivides failure into different degrees of leakage. Failure at the low-pressure end of the range would most

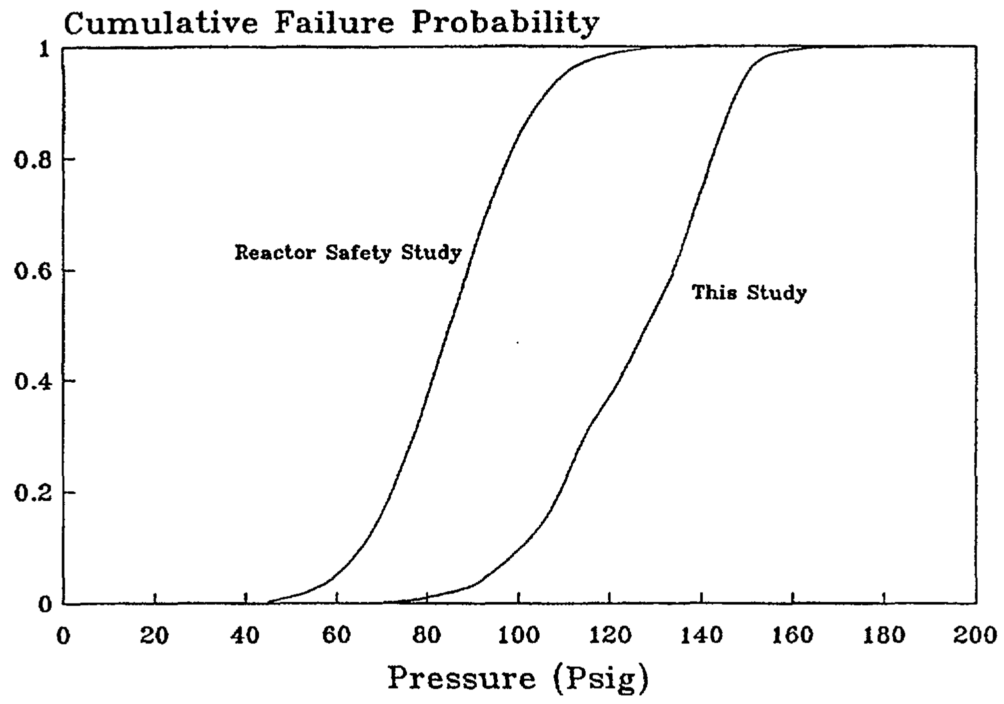


Figure 9.6 Comparison of containment failure pressure with Reactor Safety Study (Surry).

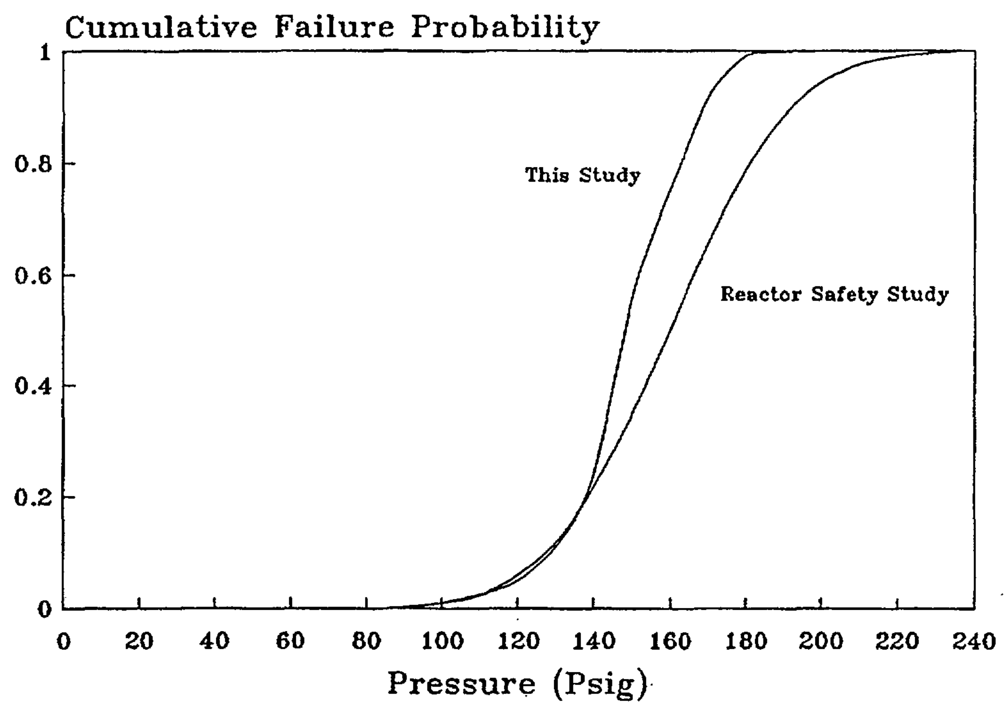


Figure 9.7 Comparison of containment failure pressure with Reactor Safety Study (Peach Bottom).

likely be the result of limited leakage, such as failure at a penetration rather than a substantial rupture of the containment wall. As the failure pressure increases, the likelihood of rupture versus leakage also increases. At pressures close to the ultimate strength of the shell, the potential for gross rupture of the containment exists but was found to be unlikely.

Figure 9.7 compares the current study with RSS estimates for cumulative failure probability as a function of pressure for the Peach Bottom plant (Mark I design). The curves are quite similar, with the current perspective being of a slightly less strong containment than the RSS representation. The curve presented from the current study is representative of a cool drywell (less than 500° F). Cumulative distributions were also developed in the current study for higher drywell temperatures. At 1200° F the median failure pressure was assessed to be 45 psig as opposed to 150 psig at low temperatures.

Failure location in the Mark I design can be as important as failure time. In the RSS, the most likely failure location was assessed to be at the upper portion of the toroidal suppression pool. It was assumed that, following containment failure, the pool would no longer be effective in scrubbing radioactive material. In the current analyses, other mechanisms of containment failure, such as direct attack of the drywell wall by molten core debris, were found to be more important than overpressure failure. The dominant location of overpressure failure is assessed to be the lifting of the drywell head by stretching the head bolts. Gases leaking past the head enter the refueling bay where limited radionuclide retention is expected rather than into the reactor building where more extensive retention could occur. (However, the leakage into the reactor building can also result in severe environments that can cause equipment failure.) Another structural failure from overpressure identified as likely in this study is at the bellows in the downcomer, which would result in leakage from the wetwell vapor space to the reactor building. Thus, although the estimated failure pressures identified in this study and in the RSS are quite similar, the modes and locations of failure are quite different.

Comparison of Surry Results

Risk in the RSS is dominated by a few key sequences for each plant. Containment performance in these sequences was a major aspect of their risk significance. The three key sequences for Surry were station blackout, an interfacing-system LOCA, and the failure of an instrumentation line penetrating the lower head. Figure 9.8 illustrates the range of early failure probability for station blackout in the current analyses and provides the point estimate from the RSS as a comparison. The RSS estimate of early failure likelihood is substantially higher than the present analysis even though the phenomenon of direct containment heating had not been identified at the time of the RSS. In addition to the lower assumed failure pressure of the containment, the RSS prediction of the rate of containment pressurization was unrealistically high.

The current perspective on the behavior of the interfacing-system LOCA in which the break occurs outside the containment resulting in bypass is essentially the same as in the RSS. The RSS did not identify the potential for rupture of a steam generator tube as a potentially important initiator of a severe accident.

The third important sequence in the RSS, involving an instrumentation line rupture, is no longer considered a core meltdown sequence. In the RSS analyses, if the containment spray injection pumps were to fail, damage was assumed to occur to the spray recirculation pumps resulting in loss of containment heat removal, containment failure, and consequent loss of emergency coolant makeup water to the vessel. More detailed analyses (Ref. 9.8) indicate, however, that condensed steam would provide sufficient water in the containment sump to prevent damage to the recirculation spray pumps, avoiding conditions resulting in containment failure and core meltdown.

Comparison of Peach Bottom Results

In the RSS analyses for the Peach Bottom plant, two sequences dominated the risk: a transient event with loss of long-term heat removal from the suppression pool and an anticipated transient without scram (ATWS). Loss of long-term heat removal is an extended accident in which heating of the suppression pool leads to overpressure failure of the containment and consequent loss of makeup water to the vessel. With the procedures now available to vent the Peach Bottom containment to outside the reactor building, the likelihood of loss of long-term heat removal leading to core meltdown has been reduced to the point where it is no longer a substantial contributor to core damage frequency or risk.

In the RSS analyses, early containment failure was considered a certainty in the ATWS sequence.

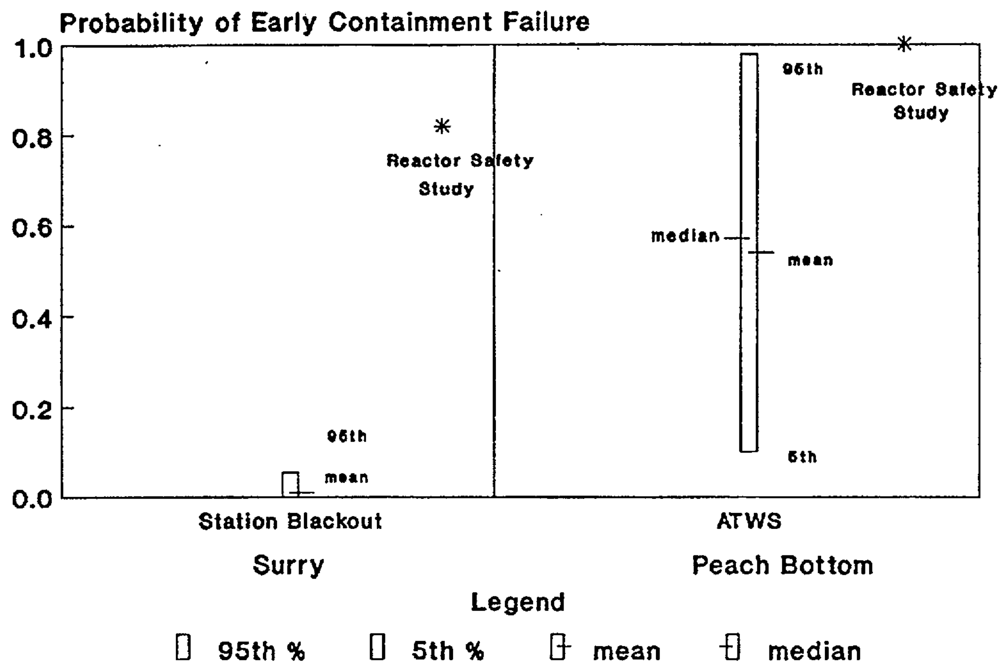


Figure 9.8 Comparison of containment performance results with Reactor Safety Study (Surry and Peach Bottom).

Figure 9.8 indicates that early failure is still considered quite likely for this sequence. The mechanisms resulting in failure and location of failure are different, however.

In summary, changes have occurred in predicting containment performance for the two plants analyzed in the RSS. There have been substantial improvements in the ability to model severe accident phenomena and system behavior in severe accidents. For Surry, the high likelihood of maintaining containment integrity indicated in the present study is the most significant difference in perspective between the two studies.

9.4 Perspectives

9.4.1 State of Analysis Methods

The analysis of severe accident loads and containment response involves substantial uncertainty because of the complexity of core meltdown processes. After a decade of research into severe accident phenomena subsequent to the TMI-2 accident, methods of analysis have been developed that are capable of addressing nearly every aspect of containment loads, including hydrogen deflagration and detonation and core-concrete interactions. In some instances, such as direct attack of the Mark I containment shell by molten material and direct containment heating, research is still being pursued (Ref. 9.9). Although the residual uncertainties are in some instances great, the methods are adequate to support meaningful Level 2 PRA analyses.

The accident progression event tree analysis techniques developed for this study involve a very detailed consideration of threats to containment integrity. A number of large computer analyses were required to support the quantification of event probabilities at each branch of the event tree. The analysis team for this study had the considerable advantage of access to researchers involved in the development and application of computer codes used in the analysis of core melt progression, core-concrete attack, containment behavior, radionuclide release and transport, and hydrogen combustion.

Computer analyses cannot, in general, be used directly and alone to calculate branching probabilities in the accident progression event tree. Since the greatest source of uncertainty is typically associated with the modeling of severe accident phenomena, the results of a single computer run (which uses a specific model) do not characterize the branching uncertainty. It is therefore necessary to use sensitivity studies, uncertainty studies, and expert judgment to characterize the likelihood of alternative events that affect the course of an accident. The effort undertaken in this study to elicit expert opinion was substantial. The expense of the overall accident progression analysis techniques (expert elicitation and computer analysis to support event tree quantification) employed in this study is currently a drawback to their widespread use. However, methods to apply the models, the distributions, and the computer codes to other plants at a reasonable cost are under study.

9.4.2 Important Mechanisms That Defeat Containment Function During Severe Accidents

The challenges to containment integrity that would occur in a severe accident depend on the nature of the accident sequence, as well as the design of the plant. The various containment designs analyzed in this study responded differently to different severe accident challenges.

Containment Bypass and Isolation Failure

When an accident occurs, a number of valves must close to isolate the containment from the environment. On the basis of absolute frequency, failure to isolate the containment was not found to be a likely source of containment failure for any of the plants analyzed. Primarily because of the low frequency of early containment failure by other means, containment isolation failure is a relatively important contributor to early failure at Zion. The subatmospheric containment and nitrogen-inerted Mark I containments are particularly reliable in this regard since it is highly likely that leakage would be identified during operation.

Containment bypass is an important contributor to large early releases of radionuclides for the Surry (subatmospheric), Sequoyah (ice condenser), and Zion (large, dry) containment designs. The principal contributors are accidents initiated by interfacing-system LOCAs and by steam generator tube ruptures. The predicted frequency of these events is quite small, however, and their dominance of risk is the result of the relatively lower frequency of other means to obtain large early releases.

Gas Combustion

Hydrogen and carbon monoxide are the two combustible gases potentially produced in large quantities in severe accidents. The principal source of hydrogen is the reduction of steam by chemical reaction of metals, particularly zirconium and iron. Carbon monoxide would only be produced in the later stages of an accident involving the attack of concrete by molten core debris. Because of the timing of carbon monoxide release, its production does not represent a threat of early failure to the containment but can contribute to delayed failure.

Rapid gas combustion was not found to be a substantial threat to containment for the Surry (subatmospheric), Zion (large, dry), or Peach Bottom (Mark I) containments. The Surry and Zion designs are sufficiently robust to survive deflagrations (rapid burning). At Surry and Zion, the likelihood of global detonations that could fail the containment (by impulsive loads) was assessed to be small. The contribution of hydrogen combustion to the pressure rise in the containment at the time of vessel failure in the event of high-pressure melt ejection of molten fuel was considered, but the likelihood of early failure of containment was also assessed to be small.

Hydrogen combustion is not a threat to the Mark I design because it normally operates with a nitrogen-inerted containment and thus has insufficient oxygen concentration to support combustion.

Hydrogen combustion was found to be a substantial threat to the integrity of the Sequoyah (ice condenser) and Grand Gulf (Mark III) designs. A very small contribution, about 1 percent, to early failure from hydrogen combustion prior to vessel breach is predicted for the station blackout sequences in Sequoyah. In arrested sequences, the containment failure probability is increased 5 percent because of ignition sources from the recovery of ac power. Approximately 12 percent mean early containment failure probability arises at the time of vessel breach, largely as the result of hydrogen combustion.

For the Grand Gulf plant, there is a substantial likelihood of containment failure before vessel breach in the short-term station blackout sequence because of the unavailability of igniters. At the time of vessel breach, hydrogen combustion loads can again occur, which can fail the containment (the percentages of containment failure before and at vessel breach are similar). Two additional reasons combine to make hydrogen events extremely important at Grand Gulf: (1) the BWR core contains an extremely large amount of zirconium that is available for hydrogen production, and (2) the suppression pool is subcooled in the short-term station blackout sequences resulting in condensation of the steam from the drywell or the vessel and leading to hydrogen-rich mixtures in the containment that are readily ignited.

Loads at Vessel Failure

The increase in containment pressure that could occur at vessel failure represents an important challenge to containment for each of the five designs (see Appendix C). In the Zion (large, dry) and Surry (subatmospheric) designs, loads at vessel breach from high-pressure melt ejections (rapid transfer of heat from dispersed core debris accompanied by chemical reactions with unoxidized metals in the debris) represent a mechanism that can result in containment loads high enough to fail containment. The predicted likelihood of failure for these scenarios in the Surry and Zion designs was found to be small, in part because most high-pressure sequences were predicted to depressurize by one or more means prior to vessel failure and because the overlap between the containment load distribution and the containment failure distribution was small.

Although loads at vessel breach have been studied more extensively for PWR containments, they were found to be an important contributor to early containment failure in the Sequoyah (ice condenser) and Peach Bottom (Mark I) plants and to early drywell failure in Grand Gulf (Mark III). In the Sequoyah and Grand Gulf analyses, hydrogen combustion is also a principal contributor to early containment failure from the loads at vessel breach. At Grand Gulf, pedestal failure, due to dynamic loads from ex-vessel steam explosions or subcompartment pressure differential, can also result in drywell failure at this stage of the accident.

Direct attack of the drywell shell is the dominant failure mechanism at vessel breach in the Peach Bottom plant. Overpressurization can also lead to leakage failure in the drywell by lifting the drywell head or to failure in the wetwell.

Direct Attack by Molten Debris

Direct attack of the drywell wall by molten debris in the Peach Bottom (Mark I) design has been the subject of considerable controversy among severe accident experts (see Section C.7 of Appendix C). Essentially half the experts whose opinions were elicited believed that containment failure would occur, and half believed that it would not occur. The numerical aggregation of these diverse views led to a mean likelihood of failure in the present analysis of approximately 30 percent when the pedestal region is wet and 80 percent when the pedestal region is dry (Ref. 9.3).

Molten debris attack was also predicted to be a threat to the Sequoyah (ice condenser containment) in high-pressure sequences in which molten debris could be dispersed into the seal table room, which is outside the crane wall and adjacent to the steel wall of the containment. The likelihood of failure was considerably less than for Peach Bottom, however.

Steam Explosions

When molten core material contacts water, the potential exists for rapid transfer of heat, production of steam, and transfer of thermal energy to mechanical work. Considerable research has been undertaken to determine the conditions under which steam explosions can occur and their energetics. At pressures near atmospheric, it is generally concluded that steam explosions would be likely if molten core material drops into a pool of water. However, the energetics and coherence of the molten fuel-coolant interaction are very uncertain. At high steam pressure, steam explosions are found to be more difficult to initiate.

Steam explosions represent a variety of potential challenges to the containment. If the interaction were to occur in the reactor vessel at the time when molten core material slumps into the lower plenum, the possibility exists of tearing loose the upper head of the vessel, which could impact and fail the containment (this has been called the "alpha mode" of containment failure since the issuance of the RSS). The analyses in this study indicate that the potential for this type of event to result in early containment failure is less than 1 percent for each of the plants. For Surry and Zion, steam explosions represent a significant fraction of the early failure probability, but only because the overall likelihood of early failure is small.

When molten core material drops into water outside the vessel, the potential failure mechanisms are different. In the Grand Gulf plant, a shock wave could propagate through water and impact the concrete structure that provides support to the reactor vessel. Substantial motion of the vessel could then lead to the tearout of penetrations through the drywell wall. Because of the shallow water pool at Peach Bottom, dynamic loads from steam explosions do not represent a similar mechanism for failures.

In addition to potentially producing missiles and shock waves, steam explosions can also rapidly generate large quantities of steam and hydrogen. The steam produced from molten fuel-coolant interactions ex-vessel following vessel breach is an important contributor to the static drywell overpressure failure in the Grand Gulf and Peach Bottom plants.

Gradual Overpressurization

Figure 9.9 illustrates the assessed pressure capability for the five plants analyzed. The ability of a containment to withstand the production of gases in a severe accident depends on the volume of the containment as well as its failure pressure. One of the principal sources of pressurization in a severe accident is steam production. In each plant design, however, engineered safety features are present to condense steam in the form of suppression pools, ice beds, sprays, air coolers, or in some designs, combinations of these systems. Steam pressurization is only a major contributor to the total pressure if, in the scenario being analyzed, the heat removal system has become inoperative; e.g., the spray system has failed, the suppression pool has become saturated, or the ice has melted.

Large quantities of hydrogen are predicted to be released in severe accidents, both in-vessel during the melting phase and ex-vessel during core-concrete attack, debris bed quenching, or high-pressure melt ejection. If the hydrogen does not burn, it will contribute to the containment pressure. Carbon monoxide and carbon dioxide produced during core-concrete attack also contribute to containment pressurization.

Because of its relatively small volume, the Peach Bottom (Mark I) design is more vulnerable to overpressurization failure by noncondensible gas generation. If the accident progression proceeds to vessel penetration and the molten core attacks the concrete, it is unlikely that containment integrity can be maintained in the long term unless other factors mitigate gas production.

Overheating

The effect of high temperature in the drywell on containment failure probability and mode was considered in the Peach Bottom analysis. Although very high gas temperatures can be achieved as the result of hydrogen combustion in the other plant designs, the structure temperatures are not predicted to reach temperatures at which the strength of the structure would be substantially reduced or sealant materials would be degraded. The Peach Bottom drywell, however, is relatively small. Substantial convective and radiative heat transfer from hot core debris could result in very high drywell wall temperatures. Failure could result from the combination of high pressure in the drywell and decreased strength of the steel containment wall. Overheating the drywell is only a contributor to scenarios in which the drywell spray is inoperative. If the sprays are operational, the drywell temperature will be much lower than for the dry case.

Drywell heating in the Peach Bottom plant represents a delayed containment failure mechanism. Since the likelihood of early failure by other mechanisms is high, drywell overtemperature failure is not a substantial contributor to risk.

Loss of Vessel Support

In the earlier section on steam explosions, a mechanism was described for drywell failure in the BWR designs in which structural failure of the reactor pedestal results in vessel motion (tipping or falling) and the tearout of piping penetrations through the drywell wall. Quasistatic pressurization of the pedestal region can result in the same phenomenon. Erosion of the pedestal by molten core attack of the concrete can also lead to the same effect. In this event, however, considerable time is required for the erosion to occur, and the failure would be late and the importance to risk is diminished. The likelihood of this mechanism of failure is generally small for the BWRs analyzed, in part because other mechanisms are likely to result in failure earlier in the accident.

Basemat Meltthrough

For each of the five plants analyzed, some potential exists for core debris to be quenched as a particulate debris bed and cooled in the reactor cavity or pedestal region if a continuous source of water is available. A significant likelihood exists, however, that, even if a replenishable water supply is available, molten core debris will attack the concrete basemat. If the core-concrete interaction does occur, the presence or absence of an overlaying water pool is not expected to have much effect on the downward progression of the melt front.

The depth of the basemat of the Peach Bottom containment, directly under the vessel, is so great that it is unlikely that the basemat would be penetrated before the occurrence of other failure modes. For the other plants, basemat penetration is possible, but the projected consequences are minor in comparison with those of aboveground failures.

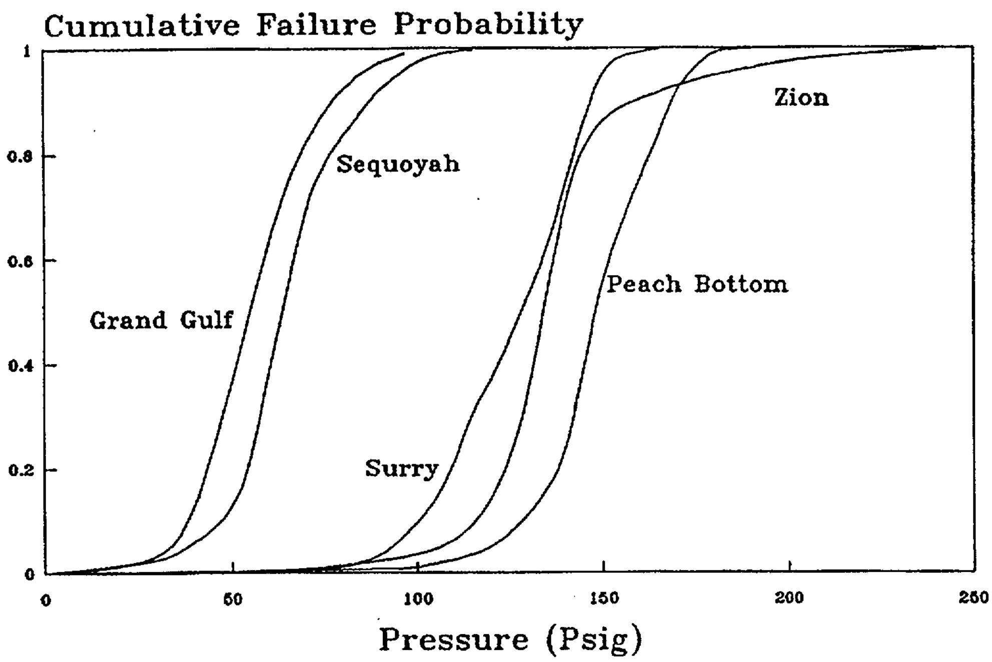


Figure 9.9 Cumulative containment failure probability distribution for static pressurization (all plants).

9.4.3 Major Sources of Uncertainty

The perspectives on the major sources of uncertainty described in this section come from four sources:

- Regression analysis-based sensitivity analyses for the mean values for risk. Simple linear regression models were used to represent the complex risk models, and adequate results were obtained. Better results would require more complex regression models. Insights for this section are deduced from the risk regression studies (regression analyses for conditional containment failure probabilities required for more detailed accident progression insights were not performed). Results of these studies are presented in References 9.2 through 9.6.

- Partial rank correlation analyses for the risk complementary cumulative distribution functions. Results of these studies are presented in References 9.2 through 9.6.

- Sensitivity studies in which separate analyses were performed with certain parameter values set to a specific value. Sensitivity studies were performed on the Mark I drywell shell meltthrough issue and the PWR RCS depressurization scenarios. These studies were only performed for the accident progression analysis; no source term or consequence insights are available.

- The subjective judgment of the analysts performing the plant-specific studies.

Importance of Accident Progression Analysis Variables to Rank Regression Analyses for Annual Risk

The majority of the variables important to the rank regression analyses performed for Surry were the initiating event frequencies of the containment bypass events and the source term variables. The only accident progression event tree variable that was demonstrated to be important to the uncertainty in risk for internal events was the probability of vessel and containment breach by an in-vessel steam explosion; this variable was moderately important to the uncertainty in total early fatality risk (Ref. 9.2).

The regression analyses performed for Sequoyah showed the containment failure pressure and loads at vessel breach to be accident progression variables somewhat important to the uncertainty in both total early fatality risk and total latent cancer fatality risk (Ref. 9.4).

The probability of drywell meltthrough was the only accident progression variable that was at all important to uncertainty in the early fatality risk or the latent cancer fatality risk for the internal regression analysis for Peach Bottom (Ref. 9.3).

The amount of hydrogen produced in-vessel, the probability of drywell failure following pedestal failure, the pressure load in the drywell at vessel breach, and the amount of hydrogen produced and released at and shortly after vessel breach were accident progression variables that were found to be important to the uncertainty in early fatality risk by the Grand Gulf regression analyses. The probability of drywell failure following pedestal failure and the pressure load in the drywell at vessel breach were found to be important to the uncertainty in latent cancer fatality risk (Ref. 9.5).

The majority of variables important to the rank regression analyses performed for Zion were related to failure or recovery of the component cooling water (CCW) system and the source term variables. The only accident progression event tree variable that was demonstrated to be important to the uncertainty in risk was the probability of vessel and containment breach by an in-vessel steam explosion. This result was also obtained from the Surry regression analyses. The probability of a steam explosion failure was found to be important to the uncertainty in both early and latent health risk measures at Zion. The importance of seal LOCA failure to risk uncertainty was expected, given the large contribution of these events to the core damage frequency. Upgrades to the Zion service water and CCW systems have the potential to reduce the importance of these events as discussed in Appendix C (Section C.15) (Ref. 9.6).

Direct Attack of Drywell Shell in Peach Bottom

The divergence of opinion of the panel of containment performance experts, in itself, is an indicator of the uncertainty in the associated phenomena. A sensitivity study was performed to determine the impact on containment performance of eliminating this failure mechanism. The mean early failure probability (averaged over all sequences) was reduced from 56 percent to 20 percent (Ref. 9.3).

High-Pressure Melt Ejection and Vessel Depressurization

For the Surry and Zion plants, early containment failure resulting from loads at vessel breach is assessed to have low probability, on the order of 1 percent. Sensitivity studies were performed to determine the dependence of this result on expert judgments made about various reactor coolant system depressurization mechanisms prior to vessel breach. A sensitivity study was performed for Surry (Ref. 9.2), which removed depressurization by temperature-induced breaks. This study indicated that removal of only temperature-induced failures for depressurization does not result in a significant increase in the likelihood of early containment failure (from roughly 1 percent to roughly 2 percent). This probability study, therefore, implies that other depressurization mechanisms, such as the failure of reactor coolant pump seals and stuck-open relief valves, are also important. However, a sensitivity study was also performed for Zion (Ref. 9.6) in which all depressurization mechanisms were removed. The result of this study was a relatively small increase in the likelihood of early containment failure. For accidents initiated by LOCAs (which dominate the estimated core damage frequency), this change resulted in essentially no change in the conditional probability of early containment failure. The probability of early failure increased by a factor of 5 for accidents initiated by transients (from roughly 0.01 to 0.06) and by a factor of 2 for accidents initiated by station blackout (from roughly 0.03 to 0.06). The reason for the relatively small impact of removing all depressurization mechanisms on the probability of early containment failure is that the Zion containment is expected to withstand high-pressure melt ejection loads (even at the upper end of the uncertainty range) with very high confidence (refer to Section C.5 of Appendix C for a more detailed discussion). Also, at these small probability levels, in-vessel steam explosions contribute to the likelihood of early containment failure. If the reactor coolant system pressure remains high, the likelihood of triggering a steam explosion is decreased. Thus, the slightly higher probability of early containment failure resulting from high-pressure melt ejection loads will be offset to some degree by the lower probability of containment failure from in-vessel steam explosions.

Uncertainties associated with high-pressure melt ejection also affect the early containment failure likelihood for the other three plants. The significance of this issue is greatest for the Sequoyah and Grand Gulf plants, which have lower overpressure capacity and which are vulnerable to the hydrogen produced in the oxidation of dispersed core debris by steam.

Containment Failure by Steam Explosions

The production of missiles by in-vessel steam explosions only appears as a significant contributor to early failure or bypass in the Zion analyses. The contribution of alpha-mode containment failure is the result of the very low probability of other modes of early failure or bypass and is itself a low value. Quasistatic and shock loading from an ex-vessel steam explosion is indicated to be a potentially important contributor to drywell failure for Grand Gulf. Ex-vessel steam explosions also contribute to quasistatic overpressurization failure in the Peach Bottom plant.

Core Melt Progression

Many of the uncertain phenomena that have the potential to lead to early containment failure (e.g., high-pressure melt ejection, drywell shell attack, steam explosions, and hydrogen generation) are sensitive to the details of core melt progression, particularly the later stages of progression in which molten core material enters the lower head of the vessel. The mass of material potentially available for dispersal at head failure, the composition of this material, the timing of head failure, and the mode of head failure have a substantial indirect impact on the likelihood of early containment failure through their effects on early failure mechanisms.

Containment Bypass

The containment bypass sequences have been discussed throughout this report as special scenarios (in which the containment function has failed) and will be briefly mentioned here. The containment bypass initiating event frequencies, transmission factors, and decontamination factors were demonstrated to be the variables most important to the uncertainty in all risk measures in both the Surry and Sequoyah rank regression analyses.

REFERENCES FOR CHAPTER 9

9.1 M. Silberberg et al., "Reassessment of the Technical Bases for Estimating Source Terms," United States Nuclear Regulatory Commission (USNRC) Report NUREG-0956, July 1986.

9.2 R. J. Breeding et al., "Evaluation of Severe Accident Risks: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 3, Revision 1, SAND86-1309, October 1990.

9.3 A. C. Payne, Jr., et al., "Evaluation of Severe Accident Risks: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4551, Vol. 4, Draft Revision 1, SAND86-1309, to be published.*

9.4 J. J. Gregory et al., "Evaluation of Severe Accident Risks: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 5, Revision 1, SAND86-1309, December 1990.

9.5 T. D. Brown et al., "Evaluation of Severe Accident Risks: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 6, Draft Revision 1, SAND86-1309, to be published.*

9.6 C. K. Park et al., "Evaluation of Severe Accident Risks: Zion Unit 1," Brookhaven National Laboratory, NUREG/CR-4551, Vol. 7, Draft Revision 1, BNL-NUREG-52029, to be published.*

9.7 USNRC, "Reactor Safety Study—An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

9.8 R. S. Denning et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios—PWR, Subatmospheric Containment Design," Battelle Columbus Division, NUREG/CR-4624, Vol. 3, BMI-2139, July 1986.

9.9 USNRC, "Revised Severe Accident Research Program Plan: Fiscal Year 1990–1992," NUREG-1365, August 1989.

*Available in the NRC Public Document Room, 2120 L Street NW., Washington, DC.

10. PERSPECTIVES ON SEVERE ACCIDENT SOURCE TERMS

10.1 Introduction

Shortly after the accident at Three Mile Island, the NRC initiated a program to review the adequacy of the methods available for predicting the magnitude of source terms for severe reactor accidents. After considerable effort and extensive peer review, the NRC published a report entitled "Reassessment of the Technical Bases for Estimating Source Terms," NUREG-0956 (Ref. 10.1). The report recommended that a set of integrated computer codes, the Source Term Code Package (STCP) (Ref. 10.2), be used as the state-of-the-art methodology for source term analysis provided that uncertainties were considered. The STCP methodology provided a starting point for source term estimates in this study. In addition, the characterization of source term uncertainties was supported by calculations with other system codes such as MELCOR (Ref. 10.3) and MAAP (Ref. 10.4), detailed special purpose codes such as CONTAIN (Ref. 10.5), as well as small codes written for this project to examine specific source term phenomena. Because it was impractical to perform an STCP calculation for each source term required and the STCP does not contain models for all potentially important phenomena, simplified methods of analysis were developed with adjustable parameters that could be benchmarked against the more detailed codes. Probability distributions, which had been developed from the elicitations of the source term panel of experts, were provided for many of the parameters in the simplified computer codes. A large number of source term estimates were generated for each plant by sampling from the probability distributions in the simplified codes.

Source terms are typically characterized by the fractions of the core inventory of radionuclides that are released to the environment, as well as the time and duration of the release, the size distribution of the aerosols released, the elevation of the release, the warning time for evacuation, and the energy released with the radioactive material. All these parameters are required for input to the MACCS (Ref. 10.6) consequence code. Although the illustrations and comparisons of source terms in this chapter emphasize the magnitude of estimated release, it is important to recognize that the other characteristics of the source term noted above, such as the timing of release, can also have an important effect on the ultimate consequences.

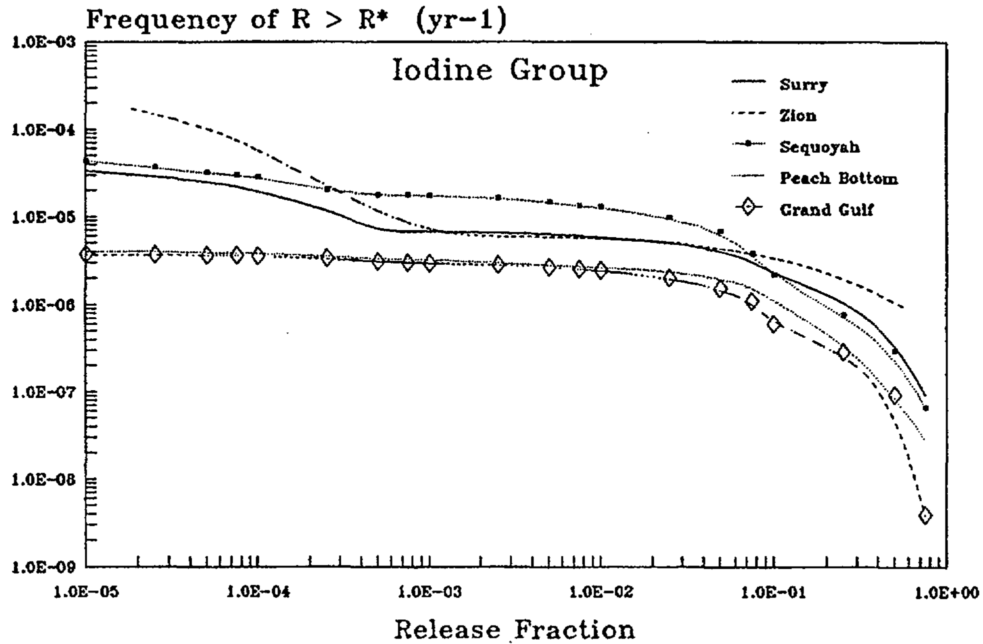
It is widely believed that the approximate treatment of source term phenomena in the Reactor Safety Study (RSS) (Ref. 10.7) analyses led to a substantial overestimation of severe accident consequences and risk. The current risk analyses provide a basis for understanding the differences that exist in source terms calculated using the new methods relative to those calculated using the RSS methods and the impact of these differences on estimated risk.

10.2 Summary of Results

Some examples of source terms (fractions of the core inventory of groups of radionuclides released to the environment) were provided for accident progression bins for each of the analyzed plants in Chapters 3 through 7. As expected, the magnitude of the source term varies between different accident progression bins depending on whether or not containment fails, when it fails, and the effectiveness of engineered safety features (e.g., BWR suppression pool) in mitigating the release. However, within an accident progression bin, which represents a specific set of accident progression events, the uncertainty in predicting severe accident phenomena is great.

In Figure 10.1, the predicted frequency of radioactive releases is compared among the five plants. In this figure, the mean distribution is presented, allowing differences in plant behavior to be illustrated. The y-coordinate in the figure represents the predicted frequency with which a given magnitude of release (the x-coordinate) would be exceeded. The location of the exceedance curve is determined by the frequencies of accident sequences in addition to the spectrum of possible source terms for those sequences.

It is not obvious in examining a radionuclide source term what the potential health impact would be to the public from a specified magnitude of release. Based on the compilation of a number of consequence analyses, however, one method (Ref. 10.8) has been developed that provides an approximate relationship for the minimum fractions of radionuclides released that result in early fatalities or early injuries. For the release of iodine, for example, the thresholds for early fatalities and early injuries occur at release fractions of the core inventory of approximately 0.1 and 0.01, respectively. Figure 10.1 does not indicate major differences in the exceedance curves for the five plant analyses. For the iodine group,



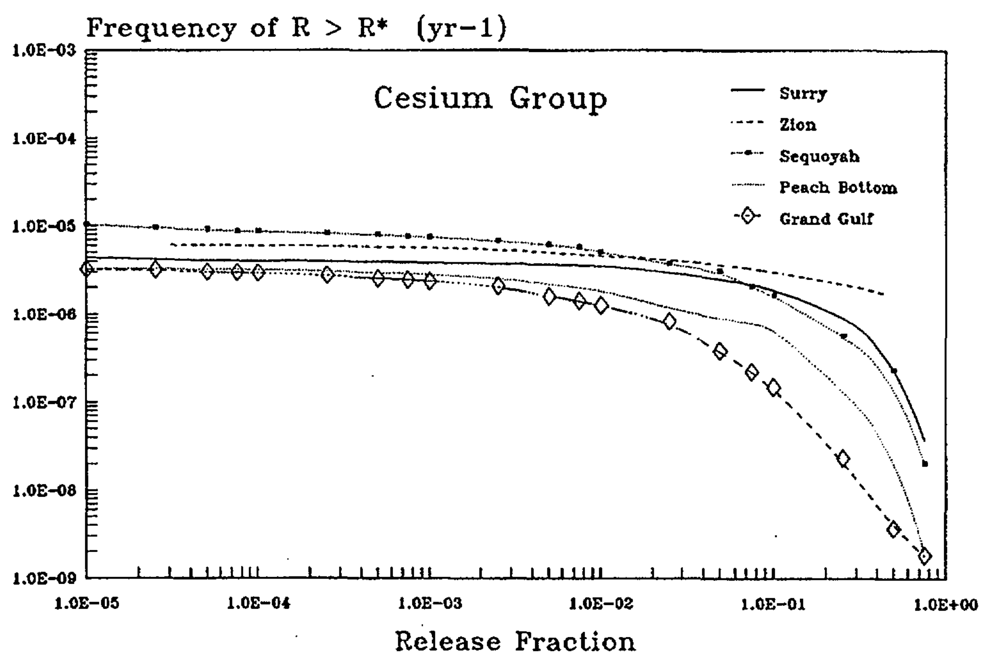
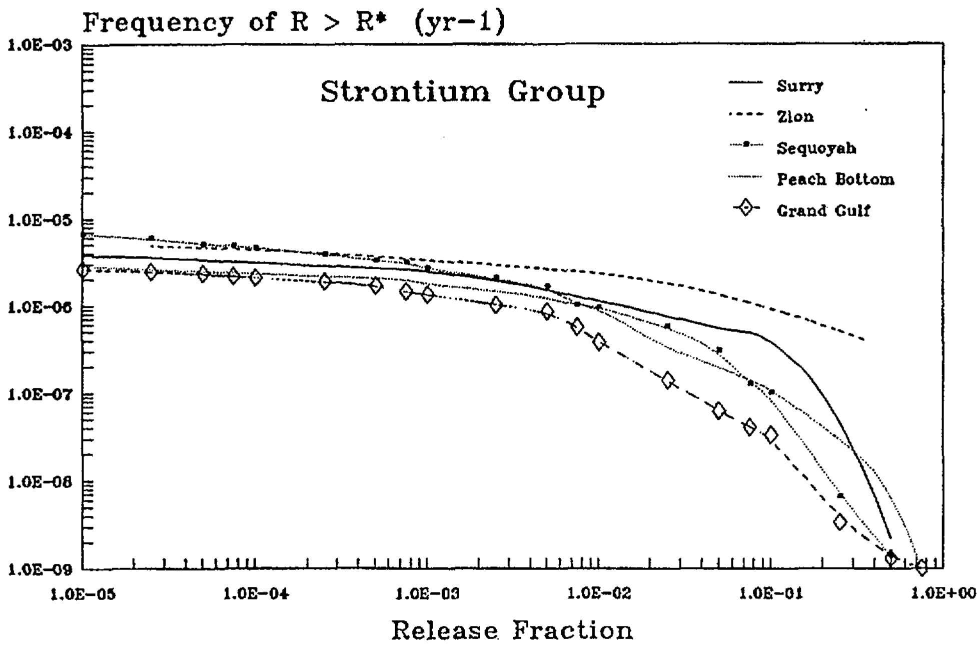


Figure 10.1 Frequency of release for key radionuclide groups.



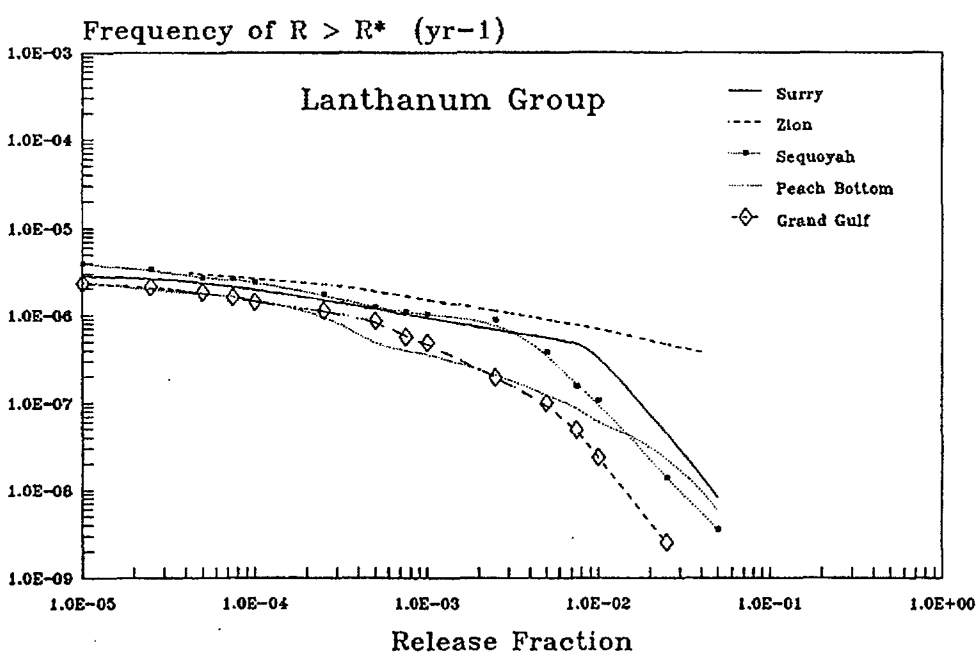


Figure 10.1 (Continued)

the frequency of exceeding a release fraction of 0.1 ranges from 1E–6 to 5E–6 per reactor year for the five plants. Similarly, for a release fraction of 0.01, the exceedance curves range from 2E–6 to 1E–5 per reactor year. The most outstanding feature of these curves is their relative flatness over a wide range of release fractions. For the iodine, cesium, and strontium groups, the curves decrease only slightly over the range of release fractions from 1E–5 to 1E–1 and then fall rapidly from 0.1 to 1. For the lanthanum group, the rapid decrease in the curve occurs at a release fraction that is approximately a decade lower. As a result of the flatness of the exceedance curves, the frequency of accidents with source terms that are marginally capable of resulting in early fatalities is only slightly less than the frequency of accidents covering a very broad spectrum of health consequences up to the occurrence of fatalities. However, the frequency of source terms with the potential for multiple early fatalities falls rapidly with increased release.

Based on the results of the source term analyses for the five plants, a number of general perspectives on severe accident source terms can be drawn:

- The uncertainty in radionuclide source terms is large and represents a significant contribution to the uncertainty in the absolute value of risk. The relative significance of source term uncertainties depends on the plant damage state.
- Source terms for bypass sequences, such as accidents initiated by steam generator tube rupture (SGTR), can be quite large, potentially comparable to the largest Reactor Safety Study source terms.
- Early containment failure by itself is not a reliable indicator of the severity of severe accident source terms. Substantial retention of radionuclides is predicted to occur in many of the early containment failure scenarios in the BWR pressure-suppression designs, particularly for the in-vessel period of release during which radionuclides are transported to the suppression pool. Containment spray system and ice condenser decontamination can also substantially mitigate accident source terms.
- Flooding of reactor cavities or pedestals can eliminate the core-concrete release of radionuclides, if a coolable debris bed is formed, or can significantly attenuate the release from the molten core-concrete interaction by scrubbing in the overlaying pool of water.

10.3 Comparison with Reactor Safety Study

In the Reactor Safety Study (RSS) (Ref. 10.7), source terms were developed for nine release categories ("PWR1" to "PWR9") for the Surry plant and five release categories for the Peach Bottom plant ("BWR1" to "BWR5"). The RSS release categories are directly analogous to the accident progression bins in the current study in that they are characterized by aspects of accident progression and containment performance that affect the source term. For example, the PWR1 release category represented early containment failure resulting from an in-vessel steam explosion with containment sprays inoperative. A point estimate for release fractions (fraction of the core inventory of an elemental group released to the environment) for seven elemental groups (in the current study, the number of elemental groups has been expanded to nine) was then used to represent this type of release.

In the current study, source terms were developed for a much larger number of accident progression bins. A distribution of release fractions was also obtained for each of the elemental groups corresponding to the individual sample members of the uncertainty analysis.

In order to simplify the presentation in this report, the results of similar accident progression bins have been aggregated to a level that is comparable to that used in the RSS. Figure 10.2 provides a comparison of an important large release category (PWR2) from the RSS for Surry with a comparable aggregation of accident progression bins (early containment failure, high reactor coolant system pressure) from the current study.* Also shown in Figure 10.2 is a low release category from the RSS (PWR7) with a comparable aggregation of accident progression bins from the current study (late failure). No range is shown for the noble gas release for this study because no permanent retention mechanisms were assumed to affect these gases. The point estimates of the release of radionuclides in the RSS early containment failure bin are more representative of the upper bounds

*Because of the aggregation of accident progression bins, some of the range of the source terms represents variation in accident progression as well as modeling uncertainty. The distribution was developed from all of the sample members within the aggregated bins without consideration of the relative frequencies of these bins.

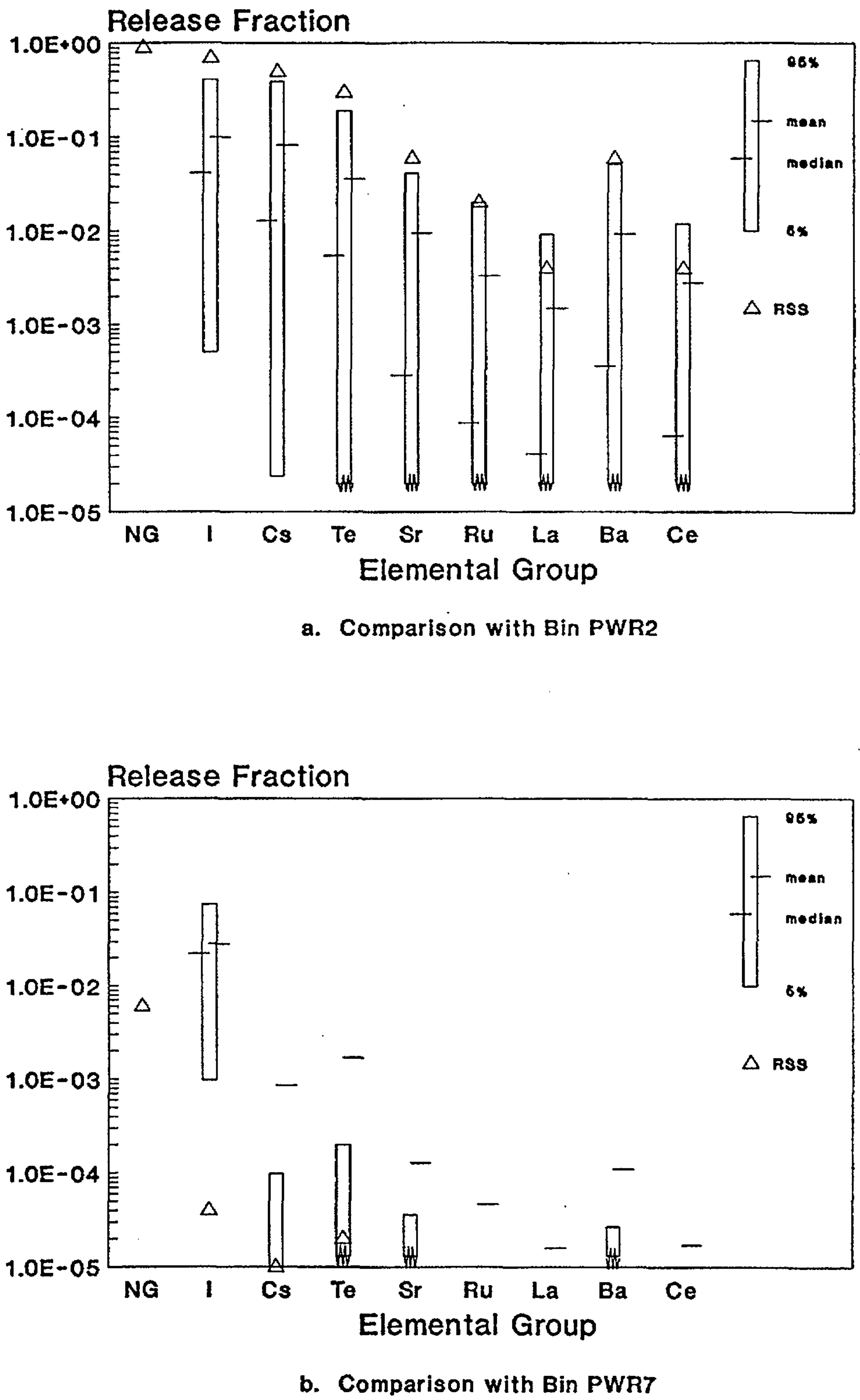


Figure 10.2 Comparison of source terms with Reactor Safety Study (Surry).

of the range in the current study than the mean or the median. For the late failure comparison, the results for this study are somewhat higher than those obtained for the RSS. The difference is related to the types of failures in the late failure bin. In the RSS, the PWR7 source terms were based on a release associated with meltthrough of the basemat in scenarios with containment sprays operable. The late failure bin in the current study also includes overpressure failure cases with a direct release from the plant to the atmosphere. Of particular significance is the nontrivial release of iodine that is associated with late release mechanisms, which were not considered in the RSS.

Figure 10.3 compares release fractions for an aggregation of early drywell failure accident progression bins from the current study with the BWR2 and BWR3 release categories. In the current study, a range of reactor building decontamination factors is considered depending on the mode of drywell failure and variations in thermal-hydraulic conditions in the building. The BWR2 release fractions are at the upper bounds of the ranges in the current study, and the BWR3 releases are near the mean values.

The second example compares results for an isolation failure in the wetwell region from the RSS, release category BWR4, with the venting accident progression bin from the current study. The RSS results are very similar to the mean release terms for the venting bin, with the exception of the iodine group, which is higher because of the late release mechanisms (reevolution from the suppression pool and the reactor vessel) considered in the current study.

Overall, the comparison indicates that the source terms in the RSS were in some instances higher and in other instances lower than those in the current study. For the early containment failure accident progression bins that have the greatest impact on risk, however, the RSS source terms appear to be larger than the mean values of the current study and are typically at the upper bound of the uncertainty range.*

10.4 Perspectives

10.4.1 State of Methods

The use of parametric source term methods, in which the parameters are fit to reproduce the results of more mechanistic codes, was found to be a practical necessity in performing a PRA that includes a complete treatment of phenomenological uncertainties. Research is in progress in some of the key areas of uncertainty that influence source term results. In a number of cases, the STCP did not have models that represent potentially important phenomena, such as revaporization from reactor coolant system surfaces and reevolution of iodine from water pools. Later codes, such as MELCOR (Ref. 10.3), which have at least rudimentary models for these processes, should provide greater assurance of consistency in the analysis. These advanced codes may not, however, remove the need for parametric codes capable of performing a large number of analyses inexpensively.

Improvement in Understanding

Since the Reactor Safety Study (RSS), substantial improvements have been made in understanding severe accident processes and source term phenomena. A major shortcoming of the RSS was the limited treatment of the uncertainties in severe accident source terms. In the intervening years, particularly subsequent to the Three Mile Island accident, major experimental and code development efforts have broadly explored severe accident behavior. In this study, care has been taken to display the assessed uncertainties associated with the analysis of accident source terms. Many of the severe accident issues that are now recognized as the greatest sources of uncertainty were completely unknown to the RSS analysts 15 years ago.

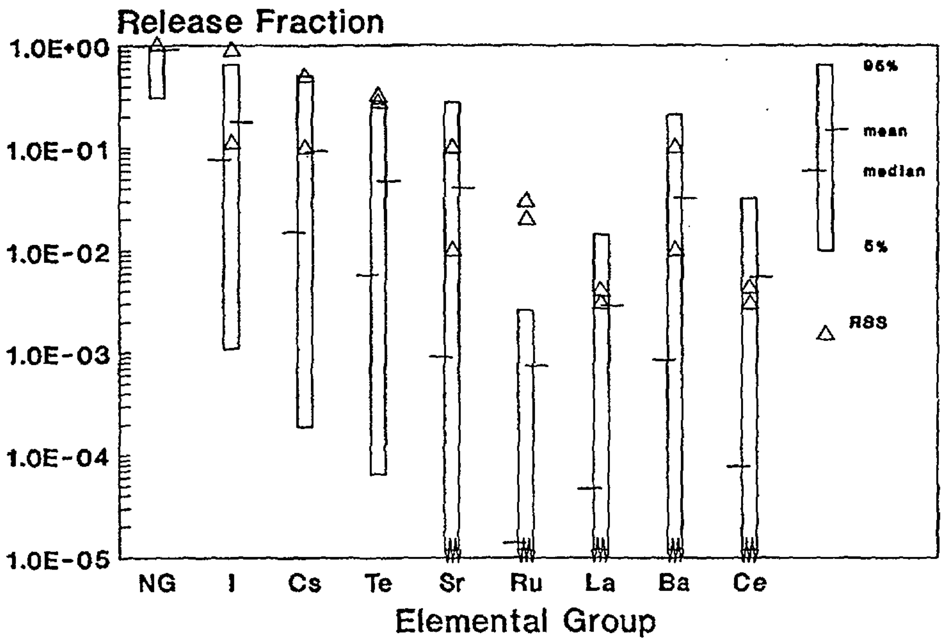
10.4.2 Important Design Features

In Chapter 9, performance of the containments of the five plants was described with respect to the timing of the onset of containment failure and the magnitude of leakage to the environment. In particular, the likelihood of early containment failure was used as a measure of containment performance. Environmental source terms are affected by more than just the mode and timing of containment failure, however. The following paragraphs describe the effect of different safety systems and plant features on the magnitude of source terms.

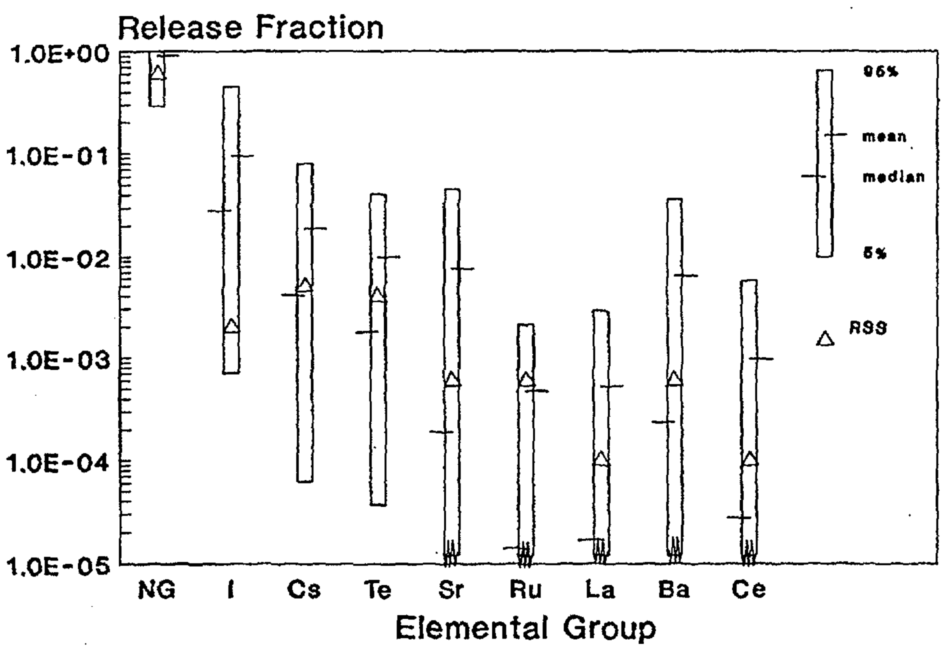
Suppression Pools

Suppression pools can be very effective in the removal of radionuclides in the form of aerosols or

*Additional comparisons with the Reactor Safety Study may be found in Reference 10.9.



a. Comparison with Bins BWR2 and BWR3



b. Comparison with Bin BWR4

Figure 10.3 Comparison of source terms with Reactor Safety Study (Peach Bottom).

soluble vapors. Some of the most important radionuclides, such as isotopes of iodine, cesium, and tellurium, are primarily released from fuel during the in-vessel release period. Because risk-dominant accident sequences in BWRs typically involve transient sequences rather than pipe breaks, the in-vessel release is directed to the suppression pool rather than being released to the drywell. As a result, the in-vessel release is subjected to scrubbing in the suppression pool, even if containment failure has already occurred. For the Peach Bottom plant, decontamination factors used in this study for scrubbing the in-vessel component ranged from approximately 1.2 to 4000, with a median value of 80. Since the early release of volatile radioactive material is typically the major contributor to early health effects, the effect of the suppression pool in depressing this component of the release is one of the reasons the likelihood of early fatalities is so low for the BWR designs analyzed.

Depending on the timing and location of containment failure, the suppression pool may also be effective in scrubbing the release occurring during core-concrete attack or reevolved from the reactor coolant system after vessel failure. In the Peach Bottom analyses, containment failure was found to be likely to occur in the drywell early in the accident. Thus, in many scenarios the suppression pool was not effective in mitigating the delayed release of radioactive material. Similarly, in the Grand Gulf design, drywell failure accompanied containment failure in approximately one-half the early containment failure scenarios analyzed. As a result, the suppression pool was found to be ineffective in mitigating ex-vessel releases in a substantial fraction of the scenarios for both BWR plants analyzed.

Although the decontamination factors for suppression pools are typically large, radioactive iodine captured in the pool will not necessarily remain there. Reevolution of iodine was found to be important in accident scenarios in which the containment has failed and the suppression pool is boiling.

Containment Sprays

If given adequate time, containment sprays can also be effective in reducing airborne concentrations of radioactive aerosols and vapors. In the Surry (subatmospheric) and Zion (large, dry) designs, approximately 20 percent of core meltdown sequences were predicted to eventually result in delayed failure or basemat meltthrough. The effect of sprays, in those scenarios in which they are operational for an extended time, is to reduce the concentration of radioactive aerosols airborne in the containment to negligible levels in comparison with non-aerosol radionuclides (e.g., noble gases) with respect to potential radiological effects. For shorter periods of operation, sprays would be less effective but can still have a substantial mitigative effect on the release.

The Sequoyah (ice condenser) design has containment sprays for the purpose of condensing steam that might bypass the ice bed, as well as for use after the ice has melted. The effects of the sprays and ice beds in removing radioactive material are not completely independent since they both tend to remove larger aerosols preferentially.

In the Peach Bottom plant, drywell sprays can be operated in sequences in which ac power is available. Scrubbing of radioactive material released from fuel during core-concrete attack can be accomplished by a water layer developed on the drywell floor, as well as by the spray droplets. Containment spray operation in Grand Gulf is most important for scenarios in which both the containment and drywell have failed. In the short-term station blackout plant damage state, power recovery that is too late to arrest core damage can still be important for the operation of containment sprays and the mitigation of the extended period of ex-vessel release from fuel.

Ice Condenser

The ice beds in an ice condenser containment remove radioactive material from the air by processes that are very similar to those in the BWR pressure-suppression pools. The decontamination factor is very sensitive to the volume fraction of steam in the flowing gas, which in turn depends on whether the air-return fans are operational. For a typical case with the air-return fans on, the magnitude of the decontamination factors was assessed to be in the range from 1.2 to 20, with a median value of 3. Thus, the effectiveness of the ice bed in mitigating the release of radioactive material is likely to be substantially less than for a BWR suppression pool.

Drywell-Wetwell Configuration

The Mark III design has the apparent advantage, relative to the Mark I and Mark II designs, of the wetwell boundary completely enclosing the drywell, in effect providing a double barrier to radioactive material release. As long as the drywell remains intact, any release of radioactive material from the fuel would be subject to decontamination by the suppression pool. For this reason, failure of the Mark III containment is not as important to severe accident risk as the potential for containment failure in combination with drywell failure. Figures 6.5 and 6.6 illustrate the difference in the environmental source terms for the early containment failure bins with and without drywell failure. With the drywell intact, the environmental source term is reduced to a level at which early fatalities would not be expected to occur, even for early failure of the outer containment. The potential advantages of the drywell-wetwell configuration were found to be limited in this study by the significant probability of drywell failure in an accident.

Cavity Flooding

The configuration of PWR reactor cavity or BWR pedestal regions affects the likelihood of water accumulation and water depth below the reactor vessel. The Surry reactor cavity is not connected by a flowpath to the containment floor. If the spray system is not operating, the cavity will be dry at vessel failure. In the Peach Bottom (Mark I) design, there is a maximum water depth of approximately 2 feet on the pedestal and drywell floor before water would overflow into the downcomer. The other three designs investigated have substantially greater potential for water accumulation in the pedestal or cavity region. In the Sequoyah design, the water depth could be as much as 40 feet.

If a coolable debris bed is formed in the cavity or pedestal and makeup water is continuously supplied, core-concrete release of radioactive material would be avoided. Even if molten core-concrete interaction occurs, a continuous overlaying pool of water can substantially reduce the release of radioactive material to the containment.

Reactor Building/Auxiliary Building Retention

Radionuclide retention was evaluated for the Peach Bottom reactor building, but an evaluation was not made for the portion of the reactor building that surrounds the Grand Gulf containment, which was assessed to have little potential for retention. The range of decontamination factors for aerosols for the Peach Bottom reactor building subsequent to drywell rupture was 1.1 to 80 with a median value of 2.6. The location of drywell failure affects the potential for reactor building decontamination. Leakage past the drywell head to the refueling building was assumed to result in very little decontamination. Failure of the drywell by meltthrough resulted in a release that was subjected to a decontamination factor of 1.3 to 90 with a median value of 4.

In the interfacing LOCA sequences in the PWRs, some retention of radionuclides was assumed in the auxiliary building (in addition to water pool decontamination for submerged releases). In the Sequoyah analyses, retention was enhanced by the actuation of the fire spray system.

Containment Venting

In the Peach Bottom (Mark I) and Grand Gulf (Mark III) designs, procedures have been implemented to intentionally vent the containment to avoid overpressure failure. By venting from the wetwell air space (in Peach Bottom) and from the containment (in Grand Gulf), assurance is provided that, subsequent to core damage, the release of radionuclides through the vent line will have been subjected to decontamination by the suppression pool.

As discussed in Chapter 8, containment venting to the outside can substantially improve the likelihood of recovery from a loss of decay heat removal plant damage state and, as a result, reduce the frequency of severe accidents. The results of this study indicate, however, only limited benefits in consequence mitigation for the existing procedures and hardware for venting. Uncertainties in the decontamination factor for the suppression pool and for the ex-vessel release and in the reevolution of iodine from the suppression pool are quite broad. As a result, the consequences of a vented release are not necessarily minor. Furthermore, the effectiveness of venting in the two plant designs is limited by the high likelihood of mechanisms leading to early containment failure, which would result in bypass of the vent.

10.4.3 Important Phenomenological Uncertainties

In order to identify the principal sources of uncertainties in the estimated risk, regression analyses were performed for each of the plant types in this study. In general, in these regression analyses, the dependent variable is risk expressed in terms of consequences per year (e.g., early fatalities per year or latent cancer fatalities per year). For the Surry plant (Ref. 10.10), however, additional regression analyses were performed in which the dependent variable is the quantity of release per year for each of the radionuclide groups. These analyses are particularly useful in investigating how uncertainties in source term variables affect the releases of different radionuclides. Also determined were partial correlation coefficients that represent the importance of uncertain variables as a function of the magnitude of the environmental release.

Relative Importance of Source Term Variables

The results of these regression analyses indicate that uncertainties in source term variables are important contributors to the uncertainties in risk but are often not the largest contributors. The relative contribution of uncertainties in source term variables depends on the characteristics of each plant damage state as illustrated in the Peach Bottom and Sequoyah regression analyses (Refs. 10.11 and 10.12). In general, the five plant analyses indicate that the importance of the aggregate of variables that affect release frequencies (accident frequency variables and accident progression variables) is similar to or greater than the importance of the aggregate of variables that affect source term magnitude.

Source term variables tend to have less importance to the uncertainty in latent cancer fatality (or population dose) risk than to the risk of early fatalities. Because of the threshold nature of early fatalities, these risk results are particularly sensitive to pessimistic values of source term variables.

Importance of Source Term Variables to Uncertainty in Environmental Release

Based on analyses performed for the Surry plant (Ref. 10.10), the importance of source term variables is seen to be different for different groups of radionuclides. The uncertainty in the release of noble gases is dominated by the uncertainty in accident frequency variables. The relative uncertainties in release fractions for the noble gases and in retention mechanisms (only volumetric holdup is assumed) are small.

The character of the risk-dominant accident sequences at Surry plays an important role in determining the importance of the source term variables for the other radionuclide groups. The steam generator tube rupture (SGTR) accident and the interfacing-system LOCA sequences (the risk-dominant sequences) involve bypass routes in which radionuclides released from the core transport to the environment without being subjected to containment deposition processes. As a result, steam generator retention and the release of radionuclides from the fuel during in-vessel melt progression are the largest contributors to uncertainty for the volatile radionuclides, iodine and cesium, and for the semivolatile radionuclides, tellurium, barium, strontium, and ruthenium. For the involatile radionuclides, lanthanum and cerium, the release of radionuclides during core-concrete interactions is also an important contributor.

The Surry analyses also indicate that the uncertainties in source term variables tend to have relatively more importance for large releases. For small releases of radionuclides, the uncertainties are dominated by the uncertainties associated with the accident frequencies.

Plant-Specific Importance of Source Term Variables to Uncertainty in Risk

Consistent with the discussion in the previous section, the largest contributors to uncertainty in early fatality risk for the Surry plant (Ref. 10.10) are the frequency of the interfacing-system LOCA sequence and two source term variables, retention in the steam generator (in an SGTR accident) and release from the fuel during in-vessel melt progression. For latent cancer fatality risk, the frequency of SGTR accidents becomes of higher importance and the frequency of interfacing-system LOCAs of reduced importance. Steam generator retention and in-vessel release of radionuclides are of comparable importance to the accident frequency variables.

The Zion results (Ref. 10.13) are similar to those for Surry but reflect a reduced significance of the interfacing-system LOCA sequence and an increased importance of steam explosions as a mode of early containment failure (this results from a much lower frequency of interfacing-system LOCA in Zion). Release of radionuclides from fuel in-vessel, steam generator retention (in an SGTR accident), and containment retention of material released prior to vessel breach (as applied in a steam explosion scenario) are the most important source term contributors to the uncertainty in early fatality risk. For latent cancer fatality risk, containment failure from a steam explosion is of reduced significance and, as a result, containment retention is not an important contributor to risk uncertainty.

For early fatality risk at Sequoyah (Ref. 10.12), the frequency of the interfacing-system LOCA is the most important contributor to uncertainty. Containment failure by overpressurization is a more likely early failure mechanism for Sequoyah than for the large, high-pressure containments at Zion and Surry. As a result, accident progression mechanisms such as pressure rise at vessel breach and containment failure pressure are also important contributors to risk uncertainty for the

Sequoyah design. The most significant source term variables are in-vessel retention fraction, containment retention fraction for the in-vessel release, and steam generator deposition (in an SGTR accident). For latent cancer fatality risk, the frequency of the SGTR accident is the most important contributor to uncertainty; none of the source term variables is significant.

Regression results were obtained for internal initiators, fire events, and seismic events for the Peach Bottom plant (Ref. 10.11). For early fatality risk from internal initiators, release from fuel in-vessel, release during core-concrete interactions, and fractional release from containment of the core-concrete source terms are the most important contributors to uncertainty. The containment building decontamination factor, late release of iodine, reactor coolant system retention, and revaporization also contribute at a level similar to the contribution from the frequencies of the accident sequences. For fire initiators, the contributions from the various source term variables are similar but slightly reduced consistent with greater uncertainty in the initiator frequency.

For latent cancer fatality risk at Peach Bottom, the important source term variables are the same as for the early fatality risk but are relatively less important than the contribution from uncertainties in the accident frequencies.

In the Grand Gulf analyses (Ref. 10.14), the source term variables were indicated to be less important than the accident sequence and accident progression variables. The most significant source term variable was indicated to be the release fraction from containment following vessel failure. The decontamination factor for the suppression pool, spray decontamination factor, in-vessel release of radioactive material, and in-vessel retention of radioactive material were also identified as moderate contributors to the uncertainty in risk.

REFERENCES FOR CHAPTER 10

10.1 M. Silberberg et al., "Reassessment of the Technical Bases for Estimating Source Terms," U.S. Nuclear Regulatory Commission (USNRC) Report NUREG-0956, July 1986.

10.2 J. A. Gieseke et al., "Source Term Code Package, A User's Guide (Mod. 1)," Battelle Columbus Division, NUREG/CR-4587, BMI-2138, July 1986.

10.3 R. M. Summers et al., "MELCOR In-Vessel Modeling," *Proceedings of the Fifteenth Water Reactor Safety Information Meeting* (Gaithersburg, MD), NUREG/CP-0091, February 1988.

10.4 Fauske and Associates, Inc., "MAAP Modular Accident Analysis Program User's Manual," Vols. I and II, IDCOR Technical Report 16.2-3, February 1987.

10.5 K. D. Bergeron et al., "User's Manual for CONTAIN 1.0, A Computer Code for Severe Reactor Accident Containment Analysis," Sandia National Laboratories, NUREG/CR-4085, SAND84-1204, July 1985.

10.6 D. I. Chanin, H. Jow, J. A. Rollstin et al., "MELCOR Accident Consequence Code System (MACCS)," Sandia National Laboratories, NUREG/CR-4691, Vols. 1-3, SAND86-1562, February 1990.

10.7 USNRC, "Reactor Safety Study—An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

10.8 G. D. Kaiser, "The Implications of Reduced Source Terms for Ex-Plant Consequence Modeling," Executive Conference on the Ramifications of the Source Term (Charleston, SC), March 12, 1985.

10.9 L. LeSage et al., "Report of the Special Committee on NUREG-1150, The NRC's Study of Severe Accident Risks," American Nuclear Society, June 1990.

10.10 R. J. Breeding et al., "Evaluation of Severe Accident Risks: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 3, Revision 1, SAND86-1309, October 1990.

10.11 A. C. Payne, Jr., et al., "Evaluation of Severe Accident Risks: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4551, Vol. 4, Draft Revision 1, SAND86-1309, to be published.*

10.12 J. J. Gregory et al., "Evaluation of Severe Accident Risks: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 5, Revision 1, SAND86-1309, December 1990.

10.13 C. K. Park et al., "Evaluation of Severe Accident Risks: Zion Unit 1," Brookhaven National Laboratory, NUREG/CR-4551, Vol. 7, Draft Revision 1, BNL-NUREG-52029, to be published.*

10.14 T. D. Brown et al., "Evaluation of Severe Accident Risks: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 6, Draft Revision 1, SAND86-1309, to be published.*

*Available in the NRC Public Document Room, 2120 L Street NW., Washington, DC.

11. PERSPECTIVES ON OFFSITE CONSEQUENCES

11.1 Introduction

Frequency distributions, in the form of complementary cumulative distribution functions (CCDFs), of four selected offsite consequence measures of the atmospheric releases of radionuclides in reactor accidents (with all source terms contributing) have been presented in Chapters 3 through 7 for the five plants* covered in this study. For each consequence measure, the 5th percentile, 50th percentile (median), 95th percentile, and the mean CCDFs were shown. This chapter provides some perspectives on the offsite consequence results for these plants.

Section 11.2 provides a discussion on the basis of the CCDFs. Section 11.3 discusses, summarizes, and compares the consequence results for the five plants displayed in the mean and the median CCDFs. Section 11.4 compares the results from the mean and median CCDFs with those of the Reactor Safety Study (Ref. 11.1). Sections 11.5 and 11.6, respectively, provide discussions on potential sources of uncertainty in consequence analysis and on sensitivities of the mean CCDFs to the assumptions on the offsite protective measures to mitigate the consequences.

Some of the perspectives provided in this chapter relate to the effectiveness of various methods of offsite emergency response. For these five plants, it appears that evacuation is the most effective emergency response for the risk-dominant accident sequences. However, as discussed below, the calculated effectiveness of a response is sensitive to assumptions on the timing of warnings to people offsite before radioactive release, the estimated delay before evacuation and the effective speed of evacuating populations, and the energy of the release. In this chapter, the results of sensitivity studies on some of these factors are discussed. The reader should not infer that these results signal a modification to NRC's emergency response guidance. Rather, they provide a glimpse of the type of technical assessment that would be required in NRC's reevaluation of emergency response.

11.2 Discussion of Consequence CCDFs

As discussed in the earlier chapters, a large number of source terms, each with its own frequency, were initially developed for each of the five plants. They spanned a wide spectrum of plant damage states, phenomenological scenarios, and source term uncertainties for each plant that led to radionuclide releases to the atmosphere. However, for the purpose of the manageability of the offsite consequence analysis, such large numbers of source terms for each plant were reduced to a much smaller number (about 30 to 60) of representative source term groups.

Each source term group was treated as a single source term in the offsite consequence analysis code, MACCS (Ref. 11.2). The MACCS analyses incorporated the mitigating effects of the offsite protective actions. The magnitudes of the selected consequence measures and their meteorology-based probabilities were calculated by MACCS for each source term group and were used to generate the meteorology-based CCDFs. These conditional CCDFs of the consequence measures for all individual source term groups served as the basic data set for further analysis. When the conditional CCDFs of a consequence measure were weighted by the frequencies of the source term groups, the 5th percentile, 50th percentile (median), 95th percentile, and the mean values of the frequencies at various magnitude levels of the consequence measure were obtained and displayed as CCDFs in Chapters 3 through 7.

Thus, in this procedure, both the frequencies of the source term groups and the probabilities of the site meteorology (which in combination with the source term groups lead to the various consequence magnitude levels) have been used in generating the percentile and mean CCDFs. (The construction of these CCDFs is discussed in Section A.9 of Appendix A.)

11.3 Discussion, Summary, and Interplant Comparison of Offsite Consequence Results

The various percentile and the mean CCDFs of the consequence measures shown in Chapters 3 through 7 display the uncertainties in the offsite consequences stemming from the in-plant uncertainties up to the source terms and their frequencies and the ex-plant uncertainties due to the variability of the site meteorology. The 5th and 95th percentile CCDFs provide a reasonable display of the bounds of the offsite consequences frequency distributions for the five plants.

*See Figures 3.9, 3.10; 4.9, 4.10; 5.8; 6.8; and 7.7, respectively, for Surry, Peach Bottom, Sequoyah, Grand Gulf, and Zion.

Tables 11.1 and 11.2 present the information contained in the mean and the median CCDFs in tabular form. Entries in these tables are the exceedance frequency levels of 10^{-5}, 10^{-6}, 10^{-7}, 10^{-8}, and 10^{-9} per reactor year and the magnitudes of the consequences that will be exceeded at these frequencies for the five plants.

As stated in Chapters 3 through 7, the CCDFs of the consequence measures presented in those chapters (and, therefore, the results shown in Tables 11.1 and 11.2) incorporate the benefits of evacuation of 99.5 percent of the population within the 10-mile plume exposure pathway emergency planning zone (EPZ), early relocation of the remaining population from the heavily contaminated areas both within and outside the 10-mile EPZ, and other protective measures. Details of the assumptions on the protective measures are presented in Table 11.3.

The results shown in Tables 11.1 and 11.2 for the five plants are discussed below.

Early Fatality Magnitudes

The early fatality magnitudes (persons) at various exceedance frequencies for a plant are driven by the core damage frequency and the radionuclide release parameters of the source term groups for the plant; the site meteorology and the population distribution in the close-in site region; and the effectiveness of the emergency response. These factors are different for the five plants. Therefore, different values of early fatality magnitudes are shown for equal levels of exceedance frequencies.

Some of the plant/site features contributing to the differences between the early fatality CCDFs of the five plants are discussed below:

- Core damage frequencies for the internal initiators for Peach Bottom and Grand Gulf are lower than those for the other three plants. Therefore, the early fatality CCDFs for Peach Bottom and Grand Gulf are associated with relatively low exceedance frequencies.

- Quantities of radionuclides associated with the early phase of the release* in the source term groups for Peach Bottom and Grand Gulf are typically smaller than those for the other three plants because of suppression pool scrubbing. This lowered the early fatality magnitudes for these two plants.

- Several source term groups for Surry and Sequoyah with large quantities of radionuclides associated with the early release phase are also associated with large thermal energy in this phase. This resulted in vertical rise of the plume in several meteorological scenarios, reducing the potential for large early fatality magnitudes.

- The time of warning before the start of the radionuclide release strongly influences the effectiveness of the emergency response, particularly the evacuation. The source term groups for Peach Bottom and Grand Gulf with potential for early fatalities, unless mitigated by emergency response, are also associated with warning times that are well in advance of the release compared to those for the other three plants because the most important accident sequences for the BWRs develop more slowly than those for the PWRs of this study. In contrast, warning times are close to the start of the release (about 40 minutes before the release) for the source term groups containing the fast-developing interfacing-system LOCA accident sequences for Surry and Sequoyah, which also have large quantities of radionuclides in the release.

- The Zion site has the highest population density within the 10-mile EPZ among the five plants (although about half of the area in this zone for Zion is water). It is followed by Surry, Sequoyah, Peach Bottom, and Grand Gulf.

- For Zion, Surry, and Sequoyah, relatively long evacuation delay times after the warnings and slow effective evacuation speeds were calculated. For Peach Bottom and Grand Gulf, relatively short evacuation delay times and fast effective evacuation speeds were calculated. Values of these parameters were based on the utility-sponsored plant-specific studies and the NRC requirements for emergency planning. The utility-sponsored evacuation time estimate studies, however, were not evaluated in terms of how well they realistically represent the sites.

In the MACCS calculations, early warnings before the radionuclide release and short evacuation

*Virtually all source term groups developed for this study have two release phases—an early release phase and a later release phase. Early fatalities are essentially due to the early release. This is because the wind direction may change before the later release, so that the later release would not always add to the radiation dose of the same people who were affected by the early release, and evacuation or relocation would likely be completed before the later release would occur.

Table 11.1 Summaries of mean and median CCDFs of offsite consequences—fatalities.

Exceedance Frequency (ry^{-1})	Early Fatalities (persons)[a]							Latent Cancer Fatalities (persons)[a]						
	1*	2*	3*	4*	5*	6*	7*	1*	2*	3*	4*	5*	6*	7*
10^{-5}														
Int.[b]	0	0	0	0	0	-	-	0	0	6(1)[c]	0	0	-	-
	0	0	0	0	0	0	0	0	0	2(1)	0	0	7(2)	1(3)
Fire	0	0	-	-	-	-	-	0	6(2)	-	-	-	-	-
	0	0	-	-	-	-	-	0	0	-	-	-	-	-
10^{-6}														
Int.	0	0	0	0	0	-	-	1(3)	1(3)	4(3)	3(2)	8(3)	-	-
	0	0	0	0	0	0	0	4(2)	2(2)	1(3)	0	2(3)	5(3)	5(3)
Fire	0	0	-	-	-	-	-	1(1)	8(3)	-	-	-	-	-
	0	0	-	-	-	-	-	7(0)	3(3)	-	-	-	-	-
10^{-7}														
Int.	3(0)	0	5(1)	0	2(2)	-	-	8(3)	8(3)	9(3)	1(3)	3(4)	-	-
	0	0	2(0)	0	2(0)	2(2)	2(0)	4(3)	3(3)	6(3)	6(2)	1(4)	2(4)	2(4)
Fire	0	0	-	-	-	-	-	4(2)	2(4)	-	-	-	-	-
	0	0	-	-	-	-	-	2(1)	1(4)	-	-	-	-	-
10^{-8}														
Int.	4(1)	0	4(2)	0	3(3)	-	-	2(4)	2(4)	2(4)	3(3)	8(4)	-	-
	0	0	5(1)	0	5(1)	1(3)	3(2)	9(3)	1(4)	1(4)	2(3)	2(4)	3(4)	3(4)
Fire	0	1(0)	-	-	-	-	-	5(3)	4(4)	-	-	-	-	-
	0	0	-	-	-	-	-	6(1)	2(4)	-	-	-	-	-
10^{-9}														
Int.	1(2)	1(0)	2(3)	0	4(3)	-	-	4(4)	4(4)	2(4)	6(3)	1(5)	-	-
	8(0)	0	2(2)	0	8(2)	4(3)	2(3)	2(4)	2(4)	2(4)	3(3)	4(4)	4(4)	5(4)
Fire	1(1)	3(0)	-	-	-	-	-	2(4)	5(4)	-	-	-	-	-
	0	0	-	-	-	-	-	1(3)	4(4)	-	-	-	-	-

*Plant Names: 1 = Surry; 2 = Peach Bottom; 3 = Sequoyah; 4 = Grand Gulf; 5 = Zion; 6 = RSS-PWR; 7 = RSS-BWR
a. First line of entries corresponds to mean CCDF; second line corresponds to median CCDF.
b. Int. = Internal initiating events
c. 6(1) = 6×10^{1} = 60

Table 11.2 Summaries of mean and median CCDFs of offsite consequences—population exposures.

Exceedance Frequency (ry^{-1})	50-Mile Region Population Exposure (person-rem)[a]					Entire Site Region Population Exposure (person-rem)[a]				
	1*	2*	3*	4*	5*	1*	2*	3*	4*	5*
10^{-5}										
Int.[b]	7(2)[c]	0	1(5)	0	5(3)	2(3)	0	4(5)	0	9(3)
	2(2)	0	4(4)	0	3(3)	3(2)	0	1(5)	0	4(3)
Fire	5(1)	1(6)	-	-	-	1(2)	3(6)	-	-	-
	0	2(3)	-	-	-	0	3(3)	-	-	-
10^{-6}										
Int.	1(6)	3(6)	3(6)	2(5)	2(7)	8(6)	7(6)	2(7)	2(6)	5(7)
	6(5)	6(5)	1(6)	1(2)	3(6)	2(6)	1(6)	7(6)	2(2)	1(7)
Fire	3(4)	1(7)	-	-	-	1(5)	5(7)	-	-	-
	2(4)	6(6)	-	-	-	6(4)	2(7)	-	-	-
10^{-7}										
Int.	8(6)	1(7)	8(6)	6(5)	8(7)	5(7)	5(7)	6(7)	9(6)	2(8)
	5(6)	6(6)	4(6)	3(5)	3(7)	2(7)	2(7)	3(7)	3(6)	7(7)
Fire	6(5)	3(7)	-	-	-	2(6)	1(8)	-	-	-
	1(5)	1(7)	-	-	-	2(5)	7(7)	-	-	-
10^{-8}										
Int.	2(7)	2(7)	2(7)	1(6)	2(8)	1(8)	1(8)	9(7)	2(7)	3(8)
	9(6)	1(7)	7(6)	6(5)	7(7)	6(7)	8(7)	6(7)	9(6)	1(8)
Fire	6(6)	5(7)	-	-	-	3(7)	2(8)	-	-	-
	5(5)	3(7)	-	-	-	6(5)	1(8)	-	-	-
10^{-9}										
Int.	3(7)	4(7)	4(7)	2(6)	4(8)	2(8)	2(8)	1(8)	3(7)	4(8)
	1(7)	2(7)	1(7)	1(6)	1(8)	1(8)	1(8)	1(8)	2(7)	2(8)
Fire	2(7)	6(7)	-	-	-	9(7)	3(8)	-	-	-
	1(6)	4(7)	-	-	-	8(6)	2(8)	-	-	-

*Plant Names: 1 = Surry; 2 = Peach Bottom; 3 = Sequoyah; 4 = Grand Gulf; 5 = Zion
a. First line of entries corresponds to mean CCDF; second line corresponds to median CCDF.
b. Int. = Internal initiating events
c. 7(2) = 7 X 10^2 = 700

Table 11.3 Offsite protective measures assumptions.

1. **Emergency Response Assumptions**

   a. Within 10-mile plume exposure pathway emergency planning zone (EPZ):

      Evacuation of people after a delay* following the warning given by the reactor operator on the imminent radionuclide release.

      Average evacuation delay times (hr): Surry 2.0, Peach Bottom 1.5, Sequoyah 2.3, Grand Gulf 1.25, Zion 2.3.

      Average effective radial evacuation speeds (mile/hr): Surry 4.0, Peach Bottom 10.7, Sequoyah 3.1, Grand Gulf 8.3, Zion 2.5.

   b. Outside of 10-mile EPZ:

      Early relocation of people: within 12 hours/24 hours after plume passage from areas where the projected lifetime effective whole body dose equivalent (EDE), as defined in ICRP Publications 26 and 30, from a 7-day occupancy would exceed 50 rems/25 rems.

      Note: These assumptions are also extended inward up to the plant site boundary for the nonevacuating or nonsheltering people.

2. **Protective Action Guides (PAGs) for Long-Term Countermeasures**

   a. FDA "emergency" PAG for directly contaminated foods and animal feeds—dose not to exceed 5-rem EDE and 15-rem thyroid (Ref. 11.3).

   b. EPA's proposed PAGs for continuation of living in contaminated environment—dose not to exceed:

      - 2-rem EDE in the first year
      - 0.5-rem EDE in the second year

      from groundshine and inhalation of resuspended radionuclides.

      Note: EPA's criteria (Ref. 11.4) are approximated in MACCS as dose not to exceed 4-rem EDE in 5 years.

   c. In absence of any Federal agency criteria for ingestion dose to an individual from foods grown on contaminated soil via root uptakes, MACCS assumes a PAG of 0.5-rem EDE and 1.5-rem thyroid for this pathway, which is similar to FDA's "preventive" PAG for directly contaminated food and animal feeds (Ref. 11.3).

*Time steps involved during the delay are: (1) notification of the offsite authorities, (2) evaluation and decision by the authorities, (3) public notification advising evacuation, and (4) people's preparation for evacuation.

delay times for Peach Bottom and Grand Gulf enabled the evacuees to have a substantial head start on the plume. This, coupled with relatively fast effective evacuation speeds, enabled the evacuees to almost always avoid the trailing radioactive plumes. Thus, the relatively lower core damage frequencies, lower magnitudes of source term groups in the early phase of release, early warnings, lower population densities, lower evacuation delays, and higher evacuation speeds made the Peach Bottom and Grand Gulf early fatality CCDFs in Figures 4.9 and 6.8 lie in the low frequency and low magnitude regions, and early fatality magnitude entries in Table 11.1 small or nil.

Surry and Sequoyah fit between Peach Bottom/ Grand Gulf and Zion. For Surry and Sequoyah, warnings close to release in the interfacing-system LOCA accident sequences made evacuation less effective for these sequences. Also, evacuation was less effective in the plume rise scenarios for those source terms for which early release phases were associated with large quantities of radionuclides and large amounts of thermal energy (sequences with early containment failure at vessel breach). With the plume rise, the highest air and ground radionuclide concentrations occur at some distance farther from the reactor (instead of occurring close to the reactor without plume rise). In such cases, the late starting evacuees from the close-in regions moving away from the reactor in the downwind direction encounter higher concentrations and receive higher doses.

Latent Cancer Fatality Magnitudes

The estimates of latent cancer fatality magnitude at various exceedance frequencies include the benefits of the protective measures discussed above. Contributions from radiation doses down to very low levels have been included. If future research concludes that it is appropriate to truncate the individual dose at a *de minimis* level, reduced latent cancer fatality estimates would be obtained.

Variations of the latent cancer fatality magnitude for the five plants at equal exceedance frequency levels primarily arise because of differences in the source term groups and their frequencies, site meteorologies, and differences in the site demography, topography, land use, agricultural practice and productivity, and distribution of fresh water bodies up to 50 to 100 miles from the plants.

Emergency response in the close-in regions has only a limited beneficial impact on delayed cancer fatality magnitude and does not contribute substantially to the differences in the cancer fatality CCDFs for the five plants. The long-term protective measures, such as temporary interdiction, condemnation, and decontamination of land, property, and foods contaminated above acceptable levels are based on the same protective action guides (PAGs) for all plants. Further, the site differences for the five plants are not large enough beyond the distances of 50 to 100 miles to contribute substantially to the differences in the latent cancer fatality CCDFs.

Population Exposure Magnitudes

Population exposure magnitudes (person-rem*) at various exceedance frequencies include the contributions from the early and chronic exposures. These magnitudes reflect the dose-saving actions of the protective measures and, therefore, are the residual magnitudes.

Variations of the population exposure magnitudes for the five plants at equal exceedance frequency levels were similar to those of the cancer fatality magnitudes discussed earlier.

The relative contributions of the exposure pathways to the population dose for a given plant are highly source term dependent. Examples of relative contributions of early and chronic exposure pathways (see Chapter 2 and Appendix A) to the meteorology-averaged mean estimates of the 50-mile and entire region population dose for selected source term groups for the five plants are shown in Table 11.4. For brevity of presentation, only four source term groups that are the top contributors to the risks of the population dose for the five plants are selected. These source term groups are designated only by their identification numbers in Table 11.4. The chronic exposure pathway is shown subdivided in terms of direct (groundshine and inhalation of resuspended radionuclides) and ingestion (food and drinking water) pathways.

For a qualitative understanding of the results shown in Table 11.4, it should be noted that:

- All radionuclides contribute to the early exposure pathway; all nonnoble gas radionuclides contribute to the chronic direct exposure pathway; and only the radionuclides of iodine, strontium, and cesium contribute to the chronic ingestion exposure pathway.

*Effective dose equivalent (EDE) (as defined in ICRP Publications 26 and 30) in the unit of rem is used in the definition of person-rem.

Table 11.4 Exposure pathways relative contributions (percent) to meteorology-averaged conditional mean estimates of population dose for selected source term groups.

Plant Name	Source Term Group Identification Number	50-Mile Region* Early Exposure	50-Mile Region* Chronic Exposure Direct	50-Mile Region* Chronic Exposure Ingestion	Entire Region* Early Exposure	Entire Region* Chronic Exposure Direct	Entire Region* Chronic Exposure Ingestion
Surry	9	28	68	2	10	69	20
	33	51	41	3	14	74	12
	37	33	58	5	9	79	12
	49	13	80	7	9	58	33
Peach Bottom	28	28	66	2	15	77	7
	34	42	47	5	24	68	5
	37	38	52	5	20	72	6
	40	23	70	3	10	81	8
Sequoyah	32	49	36	8	11	68	20
	35	42	47	6	8	59	32
	43	49	28	19	11	73	15
	44	59	29	9	12	75	13
Grand Gulf	19	24	62	12	17	46	42
	25	16	65	16	4	54	41
	28	10	72	16	3	41	57
	32	41	39	17	12	62	25
Zion	139	50	46	1	27	56	16
	175	71	21	2	49	39	8
	142	24	73	1	23	60	15
	136	44	49	2	12	67	20

*The difference between 100 percent and the sum of the pathway contributions is the relative population dose to the decontamination workers.

- Early exposure pathway population dose estimated is largely unmitigated, except for the evacuated and relocated people. In addition to cloudshine and cloud inhalation during plume passage, it includes the groundshine and inhalation of resuspended radionuclides for a period of 7 days after the radionuclide release.

- Chronic exposure pathway involves dose integration from 7 days to all future times (i.e., the sum total of the dose over time).

- In the MACCS analysis, the protective actions to mitigate the chronic exposure pathways are largely confined to the 50-mile region of the site. Outside the 50-mile region, the mitigative actions (based on the PAGs) are generally not triggered in MACCS because of the relatively low levels of contamination (however, sometimes they are triggered depending on the meteorology and the source term magnitudes).

- Protective actions are not assumed for water ingestion.

Except for Grand Gulf, Table 11.4 shows that in the 50-mile region the early exposure pathway population dose and the chronic direct exposure pathway population dose are roughly similar; the chronic ingestion pathway makes smaller contributions. For the entire region, the chronic direct exposure pathway has increased contributions relative to the early exposure pathway. This is because at longer distances the early exposure pathway has weakened as a result of low air and ground concentrations and the short (i.e., 7 days) integration time for ground exposure. Relative contributions of the chronic ingestion exposure pathway are also higher for the entire region. This is because the chronic direct exposure is dependent on population size and the chronic ingestion exposure is dependent on farmland and water body surface area. An increase in the population size with distance from a plant generally occurs less rapidly compared to the increase in the area with distance.

For Grand Gulf, generally the contributions from the early exposure pathway are lower than the chronic direct exposure pathway in the 50-mile region relative to the other four plants and are due to the characteristics of the selected source term groups. For the entire region, the relative contributions of the early exposure pathway and chronic direct exposure pathway are similar to the other plants. However, the ingestion exposure pathway has higher contributions both in the 50-mile and entire region compared to the other plants. This is because the Grand Gulf site region has a smaller population size and a larger area devoted to farming than the other four sites of this study.

11.4 Comparison with Reactor Safety Study

The mean and the median CCDFs of two of the selected consequence measures, namely, early fatalities and latent cancer fatalities, displayed in Chapters 3 through 7 for the internal initiators of the reactor accidents and summarized in Table 11.1, may be compared with the CCDFs displayed in the Reactor Safety Study (RSS). However, the RSS CCDFs are the results of superpositions of the meteorology-based conditional CCDFs for the RSS "release categories"* after being weighted by the median frequencies of the release categories. The CCDFs shown in Chapters 3 through 7 are calculated in a different way from the RSS CCDFs. Thus, they are not strictly comparable.

The RSS CCDFs of early fatalities and latent cancer fatalities are shown in the RSS Figures 5–3 and 5–5, respectively. The magnitudes of delayed cancer fatalities shown in the RSS CCDFs are actually the magnitudes of their projected uniform annual rates of occurrence over a 30-year period. Thus, these RSS rate magnitudes need to be multiplied by a factor of 30 to derive their total magnitudes. After performing this step, the RSS results have been entered in Table 11.1 for comparison with the results of this study.

Table 11.1 shows that, for one or more early fatality magnitudes, the mean and median frequencies for the three PWRs of this study (Surry, Sequoyah, and Zion) and the median frequency for the RSS-PWR are similar and are less than 10^{-6} per reactor year. However, Table 11.1 also shows that these frequencies for the two BWRs of this study (Peach Bottom and Grand Gulf) are significantly lower than that for the RSS-BWR. For one or more early fatality magnitude, the median frequency is less than 10^{-6} per reactor year for the RSS-BWR; whereas, the mean and median frequencies are less than 10^{-8} per reactor year for Peach Bottom and less than 10^{-9} per reactor year for Grand Gulf.

Further, the comparison of the early fatality magnitudes in the median exceedance frequency

*RSS "release categories" are analogous to the source term groups in the present study but were developed by different procedures.

range of 10^{-9} to 10^{-7} per reactor year shows that the RSS estimates are significantly higher than the estimates for the five plants of this study.

Table 11.1 shows that for the one or more latent cancer fatality magnitudes, the mean and median frequencies of only one plant (Sequoyah) of this study and the median frequencies for the RSS-PWR and RSS-BWR are similar and are less than 10^{-4} per reactor year. However, these frequencies for the other four plants of this study are an order of magnitude lower than that for the RSS; i.e., less than 10^{-5} per reactor year.

The RSS estimates of latent cancer fatality magnitudes for the median exceedance frequency range of 10^{-9} to 10^{-5} per reactor year are higher (in some instances significantly higher) than those for the five plants of this study—except for Zion at the median exceedance frequency of 10^{-9} per reactor year where they are about equal.

There are several factors contributing to the differences in the frequency distributions of the offsite consequences for this study and the RSS. Some of these factors are mentioned below:

- Accident sequence frequency differences.
- Source term characterization difference. Most of the source terms of this study have two releases—an early release and a later release. Early fatalities from a source term are mostly the consequences of the early release. Cancer fatalities are the consequences of both early and later releases. On the other hand, the RSS source terms did not have such a breakdown in terms of early or later release. Therefore, the early fatalities from an RSS source term were the consequences of the entire release, as were the latent cancer fatalities.
- Consequence analyses for this study are site specific, using data for the site features described in Chapters 3 through 7. The RSS consequence analysis was generic; it used composite offsite data by averaging over 68 different sites.
- In the present study, evacuation to a distance of 10 miles is assumed; whereas, in the RSS, evacuation to a distance of 25 miles was assumed.
- Health effect models of this study are different from those of the RSS.
- Protective action guide dose levels for controlling the long-term exposure are different.
- There are other miscellaneous differences between the accident consequence models and input data used in this study and the RSS.
- Different procedures were used for constructing the CCDFs.

11.5 Uncertainties and Sensitivities

There are uncertainties in the CCDFs of the offsite consequence measures. Some of these uncertainties are inherited from the uncertainties in the source term group specifications and frequencies. However, even after disregarding the source term group uncertainties, there are significant uncertainties in the CCDFs of the consequence measures due to uncertainties in the modeling of atmospheric dispersion, deposition, and transport of the radionuclides; transfer of radionuclides in the terrestrial exposure pathways; emergency response and long-term countermeasures; dosimetry, shielding, and health effects; and uncertainties in the input data for the model parameters.

Because of time constraints, uncertainty analyses for the offsite consequences, except for the uncertainties due to variability of the site meteorology, have not been performed for this report. They are planned for future studies. For this study, only best estimate values of the parameters for representation of the natural processes have been used in MACCS. An analysis of sensitivity of the CCDFs to the alternative protective measure assumptions is provided in the following section.

11.6 Sensitivity of Consequence Measure CCDFs to Protective Measure Assumptions

Emergency response, such as evacuation, sheltering, and early relocation of people, has its greatest beneficial impact on the early fatality frequency distributions. The long-term protective measures, such as decontamination, temporary interdiction, and condemnation of contaminated land, property, and foods in accordance with various radiological protective action guides (PAGs), have their largest beneficial impact on the latent cancer fatality and population exposure frequency distributions.

11.6.1 Sensitivity of Early Fatality CCDFs to Emergency Response

Four alternative emergency response modes within the 10-mile EPZ, as characterized in Table

11.5, are assumed in order to show the sensitivity of early fatality CCDFs to these response modes.

Table 11.6 summarizes the early fatality mean CCDFs in tabular form for Surry, Peach Bottom, Sequoyah, and Grand Gulf for two alternative emergency response modes, and Zion for all four alternative emergency response modes. Several inferences are drawn later in this section regarding the effectiveness of these alternative emergency response modes for the five plants based on these data. However, more analysis is needed to support these inferences for emergency response and to provide detailed insight into the underlying competing processes involved that diminish or enhance the effectiveness of any emergency response mode.

In particular, the effectiveness of evacuation is very site specific and source term specific. It is largely determined by two site parameters, namely, evacuation delay time and effective evacuation speed, and two source term parameters—warning time before release and energy associated with the release (which, during some meteorological conditions, could cause the radioactive plume to rise while being transported downwind). Therefore, it cannot be extrapolated across the source terms for a plant or across the plants for similar source terms.

The CCDFs discussed here include contributions from many source term groups. The effectiveness of any emergency response mode judged from the sensitivity of the early fatality mean CCDF for a plant is essentially the effectiveness for the dominant source terms in specific frequency intervals included in the CCDF. With these caveats, the inferences based on the data shown in Table 11.6 are as follows:

Zion

1. Evacuation from the 0-to-5 mile EPZ combined with sheltering in the 5-to-10 mile EPZ is as effective as evacuation from the entire 10-mile EPZ. Effectiveness of evacuation in close-in regions of radius less than 5 miles and sheltering in the outer regions will be evaluated in future studies. (See Chapter 13.)

2. Sheltering, due to better shielding protection indoors, is more effective than early relocation from the state of normal activity. (See Tables 11.3 and 11.5 for distinctions between evacuation, early relocation, and sheltering modes of response assumed in this study.)

Sequoyah

1. Evacuation is more effective than relocation for exceedance frequencies higher than 10^{-8} per reactor year.

2. In the low frequency region (i.e., 10^{-8} per reactor year or less), the early relocation mode is more effective than evacuation. This "crossover" of the early fatality mean CCDFs for the two response modes is likely because of the dominance of the low frequency large source terms that also have short warning times before release and/or high energy contents and calculated long evacuation delay time and slow effective evacuation speed. Because of the short warning time before release and a long delay between the warning and the start of evacuation, many evacuees become vulnerable to the radiation exposures from the passing plume and contaminated ground rather than escape these exposures. Because of the plume-rise effect (for the hot plumes), the peak values of the air and ground radionuclide concentrations occur at some distance farther from the plant. In such a case, the evacuees from close-in regions moving in the downwind direction move from areas of lower concentrations to areas of higher concentrations and receive a higher dose. It should be noted that, while evacuating, the people are out in the open and have minimal shielding protection. For the above situations, the sheltering mode also would show the same crossover effect.

   However, the crossover effect showing that relocation or sheltering may be more effective than evacuation may not be realistic because of uncertainties in the consequence analysis.

Peach Bottom, Grand Gulf

The source terms and features of these two low population density sites make evacuation a very effective mode of offsite response.

Surry

Although entries in Table 11.6 show that evacuation is more effective than relocation from the state of normal activity, some low probability accident sequences for Surry are similar to those of Sequoyah (short warning times of the interfacing-system LOCA accident sequences and large

Table 11.5 Assumptions on alternative emergency response modes within 10-mile plume exposure pathway EPZ for sensitivity analysis.

a. Evacuation (see Table 11.3).

b. Early relocation in lieu of evacuation or shelter: Extends the assumptions for relocation outside the 10-mile EPZ (see Table 11.3) inward up to the plant site boundary.

c. Sheltering* (getting to and remaining indoors) in lieu of evacuation, followed by fast relocation after plume passage.

d. Evacuation for the inner 0-5 mile region and sheltering* in the outer 5-10 mile region followed by fast relocation after plume passage.

*<u>Sheltering assumptions details</u>: After an initial delay of 45 minutes from the reactor operator's warning, people get indoors and remain indoors and are relocated to uncontaminated areas within a maximum of 24 hours of remaining indoors. However, virtually all source terms analyzed in this study have two release phases—an early (first) release and a later (second) release. If there is a sufficient time gap (about 4 hours) between the two release phases, then people from indoors can be relocated to uncontaminated areas during this gap and avoid the exposure from the second release. With this perspective, two cases of relocation earlier than 24 hours are implemented in calculations as follows:

- Relocation within 4 hours after termination of the initial (the first) release, if the second release does not occur within this 4 hours; otherwise,
- Relocation within 4 hours after termination of the second release (provided this relocation time is earlier than 24 hours of indoor occupancy; otherwise, relocation is at 24 hours of indoor occupancy).

The dose for the above extra 4-hour period is assumed to account for the dose during the period of waiting for the plume to leave the area after termination of the release and the dose during people's transit to the relocation areas.

Table 11.6 Sensitivity of mean CCDF of early fatalities to assumptions on offsite emergency response.

Exceedance Frequency (ry^{-1})	10-mile EPZ Emergency Response Mode*	Early Fatalities (persons)				
		Surry	Peach Bottom	Sequoyah	Grand Gulf	Zion
10^{-5}	a. Evacuation	0/0	0/0	0	0	0
	b. Relocation	0/0	0/0	0	0	0
	c. Shelter	**	**	**	**	0
	d. Evac/Shelter	**	**	**	**	0
10^{-6}	a. Evacuation	0/0	0/0	0	0	0
	b. Relocation	0/0	0/2(1)	6(0)	0	6(0)[a]
	c. Shelter	**	**	**	**	0
	d. Evac/Shelter	**	**	**	**	0
10^{-7}	a. Evacuation	0/0	0/0	5(1)	0	2(2)
	b. Relocation	2(1)/0	1(1)/1(2)	7(1)	2(0)	1(3)
	c. Shelter	**	**	**	**	7(2)
	d. Evac/Shelter	**	**	**	**	2(2)
10^{-8}	a. Evacuation	4(1)/0	0/0	4(2)	0	3(3)
	b. Relocation	2(2)/0	7(1)/3(2)	2(2)	2(1)	8(3)
	c. Shelter	**	**	**	**	6(3)
	d. Evac/Shelter	**	**	**	**	3(3)
10^{-9}	a. Evacuation	1(2)/1(1)	0/0	2(3)	0	4(3)
	b. Relocation	9(2)/5(1)	2(2)/5(2)	6(2)	8(1)	2(4)
	c. Shelter	**	**	**	**	9(3)
	d. Evac/Shelter	**	**	**	**	4(3)

Note: Under each plant name, the first entry is for the internal initiators and the second entry is for fire.

*See Table 11.3 for assumptions.
**No data
a. $6(0) = 6x10^0 = 6$

thermal energy for the sequences with early containment failure at vessel breach). Analyses of the sensitivity of early fatality CCDFs to sheltering, or a combination of evacuation and sheltering, have not been performed for Surry (nor for Peach Bottom, Sequoyah, and Grand Gulf).

11.6.2 Sensitivity of Latent Cancer Fatality and Population Exposure CCDFs to Radiological Protective Action Guide (PAG) Levels for Long-Term Countermeasures

The potential for latent cancer fatalities and population exposure is assumed to exist down to any low level of radiation dose and, therefore, over the entire site region. Although both early and chronic exposure pathways contribute to these consequence measures, only the chronic exposure pathways are expected to be mitigated by the long-term countermeasures such as decontamination, temporary interdiction, or condemnation of contaminated land, property, and foods based on guidance provided by responsible Federal agencies in terms of PAGs. This implies that, if the radiation dose to an individual from a chronic exposure pathway would be projected to exceed the PAG (or intervention) level for that pathway, countermeasures should be undertaken to reduce the projected dose from the pathway so that it does not exceed the PAG level. Therefore, the latent cancer fatalities and the population exposures stemming from the chronic exposure pathways are expected to be sensitive to the PAG values.

The chronic exposure pathways base case PAGs are shown in Table 11.3. The only alternative PAG used for this sensitivity analysis is the RSS PAG for the groundshine dose to an individual for continuing to live in the contaminated environment. The RSS PAG adopted here is 25-rem EDE from groundshine and inhalation of resuspended radionuclides (instead of the RSS 25-rem whole body dose from groundshine only) in 30 years. This alternative is used to replace the base case PAG of 4-rem EDE in 5 years.

Summaries of the latent cancer fatality and population exposure mean CCDFs for both cases for the five plants for the internal initiating events are shown in Table 11.7.

Table 11.7 shows that there is practically no difference between the consequence magnitudes for the five plants for the two PAGs for continuing to live in the contaminated environment at the exceedance frequency of 10^{-5} per reactor year. This is because the source terms with frequency 10^{-5} per reactor year or higher have low release magnitudes such that the resulting environmental contaminations are below both the EPA and RSS PAG-based trigger levels for protective actions (i.e., no protective actions are needed).

At lower exceedance frequencies, source terms with larger release magnitudes contribute and the two PAGs reduce the consequences to different extents. The RSS PAG is less restrictive than the EPA PAG. Thus, the long-term consequence magnitudes with the RSS PAG are generally higher than those with the EPA PAG at equal exceedance frequencies. However, the economic consequences, discussed in the supporting contractor reports (Refs. 11.5 through 11.9), would show just the opposite behavior, i.e., economic consequences would be higher for the EPA PAG than for the RSS PAG.

Table 11.7 Sensitivity of mean CCDFs of latent cancer fatalities and population exposures to the PAGs for living in contaminated areas—internal initiating events.

Exceedance Frequency (ry-1)	Cancer Fatalities (persons)					50-Mile Pop. Exp. (person-rem)					Entire Region Pop. Exp. (person-rem)				
	1*	2*	3*	4*	5*	1*	2*	3*	4*	5*	1*	2*	3*	4*	5*
10^{-5}															
EPA+	0	0	6(1)[a]	0	0	7(2)	0	1(5)	0	5(3)	2(3)	0	4(5)	0	9(3)
RSS+	0	0	6(1)	0	0	7(2)	0	1(5)	0	5(3)	2(3)	0	4(5)	0	9(3)
10^{-6}															
EPA	1(3)	1(3)	4(3)	3(2)	8(3)	1(6)	3(6)	3(6)	2(5)	2(7)	8(6)	7(6)	2(7)	2(6)	5(7)
RSS	2(3)	2(3)	5(3)	3(2)	1(4)	2(6)	4(6)	5(6)	2(5)	3(7)	1(7)	1(7)	3(7)	2(6)	8(7)
10^{-7}															
EPA	8(3)	8(3)	9(3)	1(3)	3(4)	8(6)	1(7)	8(6)	6(5)	8(7)	5(7)	5(7)	6(7)	9(6)	2(8)
RSS	9(3)	1(4)	1(4)	2(3)	4(4)	1(7)	2(7)	1(7)	1(6)	2(8)	6(7)	7(7)	6(7)	1(7)	2(8)
10^{-8}															
EPA	2(4)	2(4)	2(4)	3(3)	8(4)	2(7)	2(7)	2(7)	1(6)	2(8)	1(8)	1(8)	9(7)	2(7)	3(8)
RSS	2(4)	4(4)	2(4)	4(3)	1(5)	2(7)	4(7)	2(7)	2(6)	3(8)	2(8)	2(8)	1(8)	2(7)	4(8)
10^{-9}															
EPA	4(4)	4(4)	2(4)	6(3)	1(5)	3(7)	4(7)	4(7)	2(6)	4(8)	2(8)	2(8)	1(8)	3(7)	4(8)
RSS	5(4)	4(4)	3(4)	6(3)	-	4(7)	6(7)	4(7)	3(6)	4(8)	3(8)	5(8)	2(8)	4(7)	4(8)

* Plant Names: 1 = Surry; 2 = Peach Bottom; 3 = Sequoyah; 4 = Grand Gulf; 5 = Zion
+ Long-term relocation PAGs:
EPA = 4-rem EDE in 5 years from groundshine—an approximation of EPA-proposed long-term relocation PAG
RSS = 25-rem EDE in 30 years from groundshine—RSS long-term relocation PAG
a. 6(1) = 6×10^1 = 60

REFERENCES FOR CHAPTER 11

11.1 U.S. Nuclear Regulatory Commission,"Reactor Safety Study—An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

11.2 D. I. Chanin, H. Jow, J. A. Rollstin et al., "MELCOR Accident Consequence Code System (MACCS)," Sandia National Laboratories, NUREG/CR-4691, Vols. 1-3, SAND86-1562, February 1990.

11.3 U.S. Department of Health and Human Services/Food and Drug Administration, "Accidental Radioactive Contamination of Human Food and Animal Feeds; Recommendations for State and Local Agencies," *Federal Register*, Vol. 47, No. 205, pp. 47073-47083, October 22, 1982.

11.4 U.S. Environmental Protection Agency, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," Draft, 1989.

11.5 R. J. Breeding et al., "Evaluation of Severe Accident Risks: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 3, Revision 1, SAND86-1309, October 1990.

11.6 A. C. Payne, Jr., et al., "Evaluation of Severe Accident Risks: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4551, Vol. 4, Draft Revision 1, SAND86-1309, to be published.*

11.7 J. J. Gregory et al., "Evaluation of Severe Accident Risks: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 5, Revision 1, SAND86-1309, December 1990.

11.8 T. D. Brown et al., "Evaluation of Severe Accident Risks: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 6, Draft Revision 1, SAND86-1309, to be published.*

11.9 C. K. Park et al., "Evaluation of Severe Accident Risks: Zion Unit 1," Brookhaven National Laboratory, NUREG/CR-4551, Vol. 7, Draft Revision 1, BNL-NUREG-52029, to be published.*

*Available in the NRC Public Document Room, 2120 L Street NW., Washington, DC.

12. PERSPECTIVES ON PUBLIC RISK

12.1 Introduction

One of the objectives of this study has been to gain and summarize perspectives regarding risk to public health from severe accidents at the five studied commercial nuclear power plants. In this chapter, risk measures for these plants are compared and perspectives drawn from these comparisons.

As discussed in Chapter 2, the quantitative assessment of risk involves combining severe accident sequence frequency data with corresponding containment failure probabilities and offsite consequence effects. An important aspect of the risk estimates in this study is the explicit treatment of uncertainties. The risk information discussed here includes estimates of the mean and the median of the distributions of the risk measures and the 5th percentile and the 95th percentile values. The risk results obtained have been analyzed with respect to major contributing accident sequences, plant-specific design and operational features, and accident phenomena that play important roles.

The assessments of plant risk that support the discussions of this chapter are discussed in detail in References 12.1 through 12.7 and summarized in Chapters 3 through 7 for the five individual plants. Appendix C to this report provides more detailed information on certain technical issues important to the risk studies. This work was performed by Sandia National Laboratories (on the Surry, Sequoyah, Peach Bottom, and Grand Gulf plants) and Idaho National Engineering Laboratory and Brookhaven National Laboratory (on the Zion plant).

12.2 Summary of Results

Estimates of risk presented in Chapters 3 through 7 for the five plants studied are compared in this section. Risk measures that are used for these comparisons are: early fatality, latent cancer fatality, average individual early fatality, and average individual latent cancer fatality risks for internally initiated and externally initiated (fire) events (additional risk measures are provided in Refs. 12.3 through 12.7). For reasons discussed in Chapter 1, seismic risk is not discussed here.

In order to display the variabilities in the noted risk measures, the early fatality and latent cancer fatality risk results of all five plants from internally initiated accidents are plotted together in Figure 12.1. Individual early fatality and latent cancer fatality risks from internally initiated accidents are compared with the NRC safety goals* (Ref. 12.8) in Figure 12.2. Similar risk results from externally initiated (fire) accidents for the Surry and Peach Bottom plants are presented in Figures 12.3 and 12.4. Estimates of the frequencies of a "large release" of radioactive material (using a definition of large as a release that results in one or more early fatalities) are presented in Figure 12.5.

Based on the results of the risk analyses for the five plants, a number of general conclusions can be drawn:

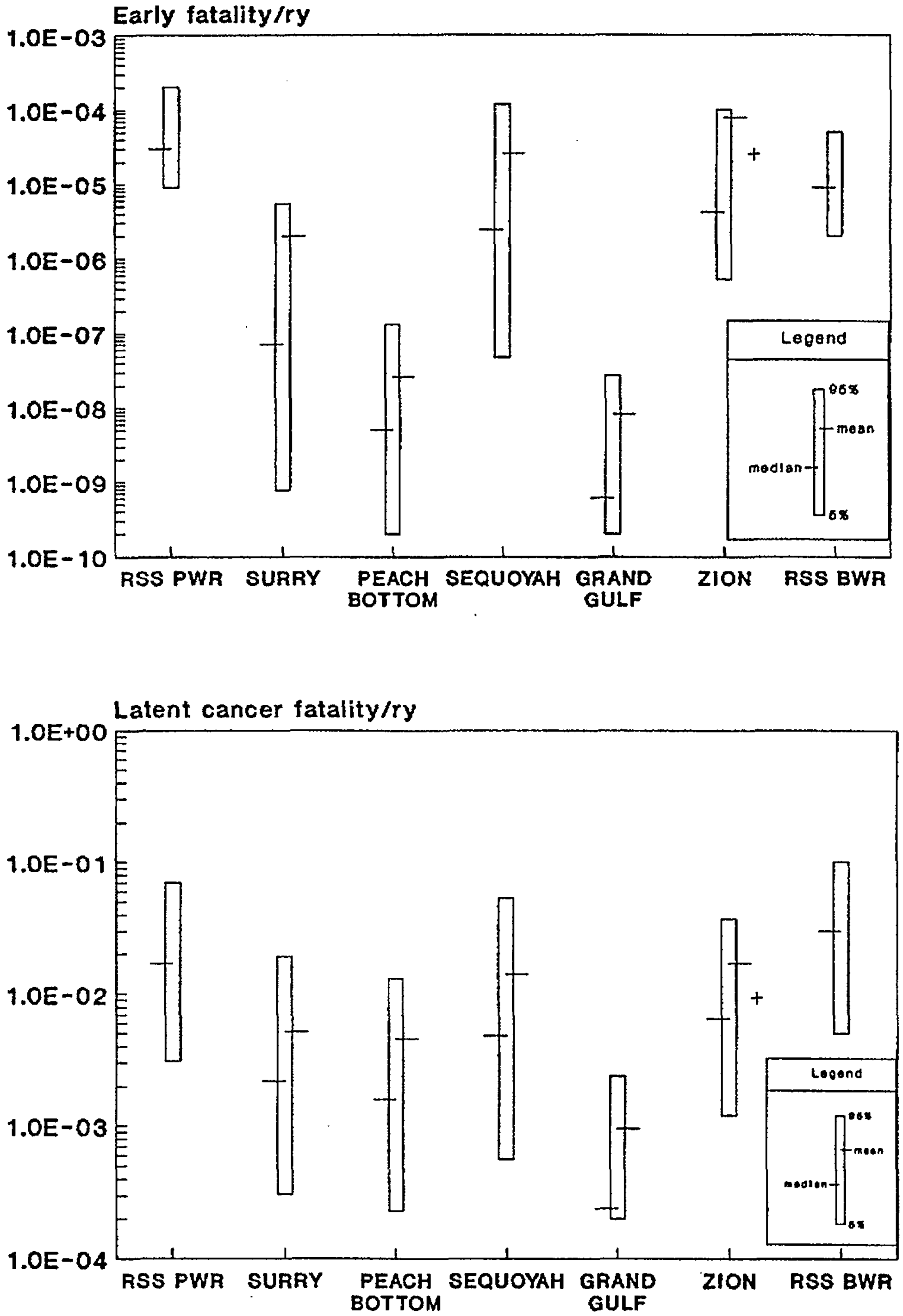
- The risks to the public from operation of the five plants are, in general, lower than the Reactor Safety Study (Ref. 12.10) estimates for two plants in 1975. Among the five plants studied, the two BWRs show lower risks than the three PWRs, principally because of the much lower core damage frequencies estimated for these two plants, as well as the mitigative capabilities of the BWR suppression pools during the early portions of severe accidents.

- Individual early fatality and latent cancer fatality risks from internally initiated events for all of these five plants, and from fire-initiated accidents for Surry and Peach Bottom, are well below the NRC safety goals.

- Fire-initiated accident sequences have relatively minor effects on the Surry plant risk compared to the risks from internal events but have a significant impact on Peach Bottom risk.

- The Surry and Zion plants benefit from their strong and large containments and therefore have lower conditional early containment failure probabilities. The Peach Bottom and Grand Gulf have higher conditional probabilities of early failure, offsetting to some degree the risk benefits of estimated lower core damage frequencies for these plants.

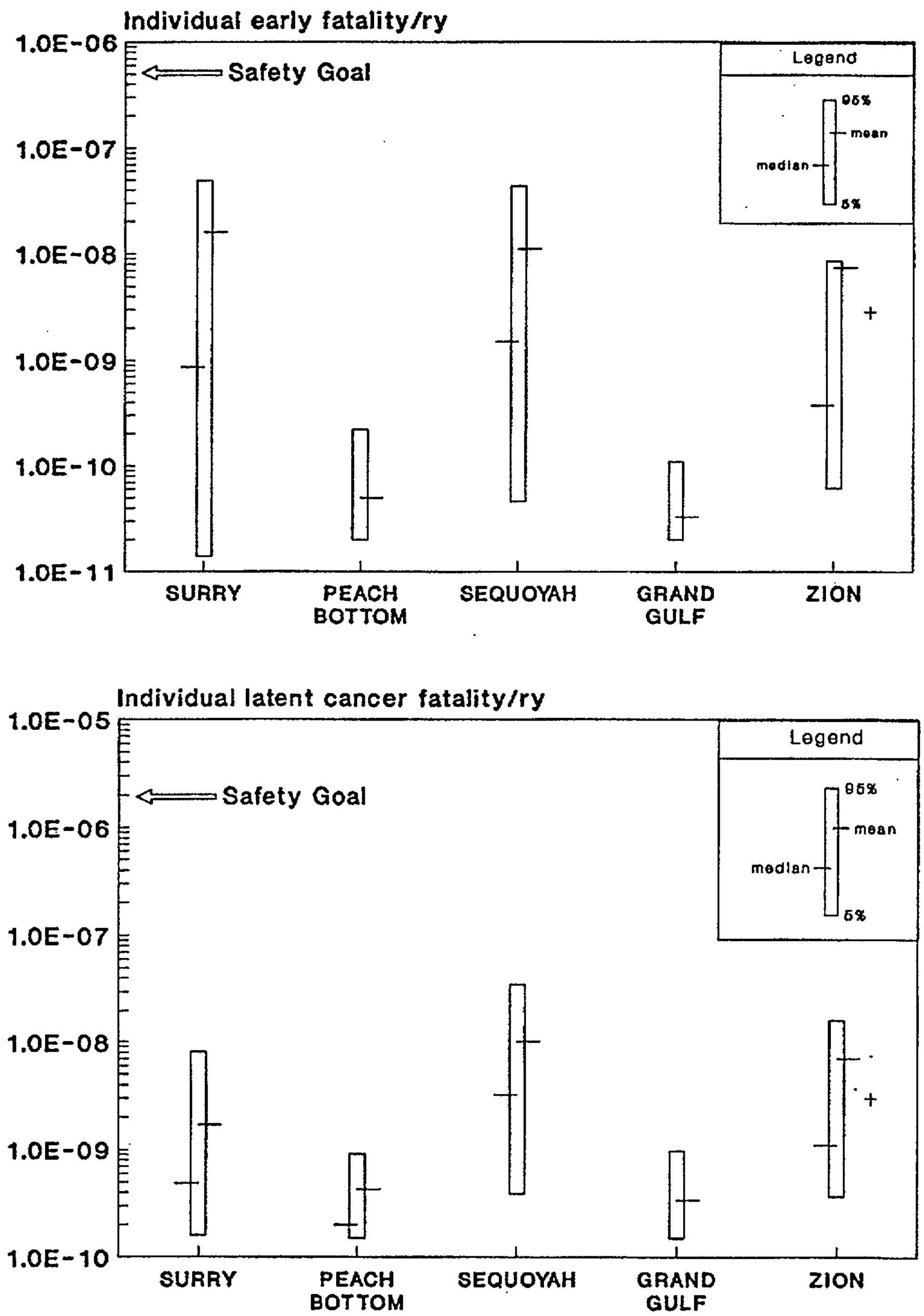
*Throughout this report, discussion of and comparison with the NRC safety goals relates specifically and only to the two quantitative health objectives identified in the Commission's policy statement (Ref. 12.8).



Notes: As discussed in Reference 12.9, estimated risks at or below 1E-7 should be viewed with caution because of the potential impact of events not studied in the risk analyses.

"+" indicates recalculated mean value based on recent modifications to the Zion plant (as discussed in Section C.15).

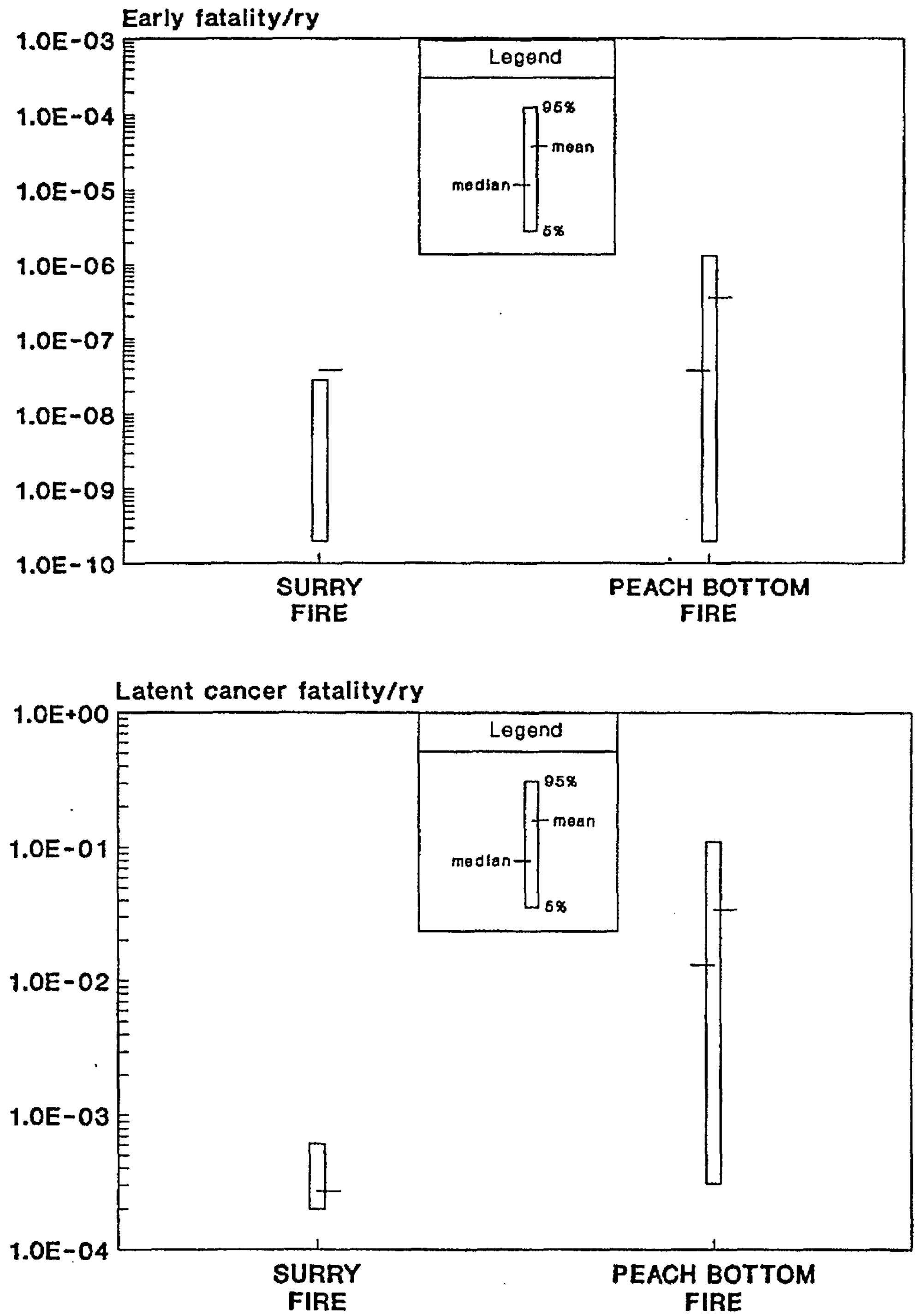
Figure 12.1 Comparison of early and latent cancer fatality risks at all plants (internal events).



Notes: As discussed in Reference 12.9, estimated risks at or below 1E-7 should be viewed with caution because of the potential impact of events not studied in the risk analyses.

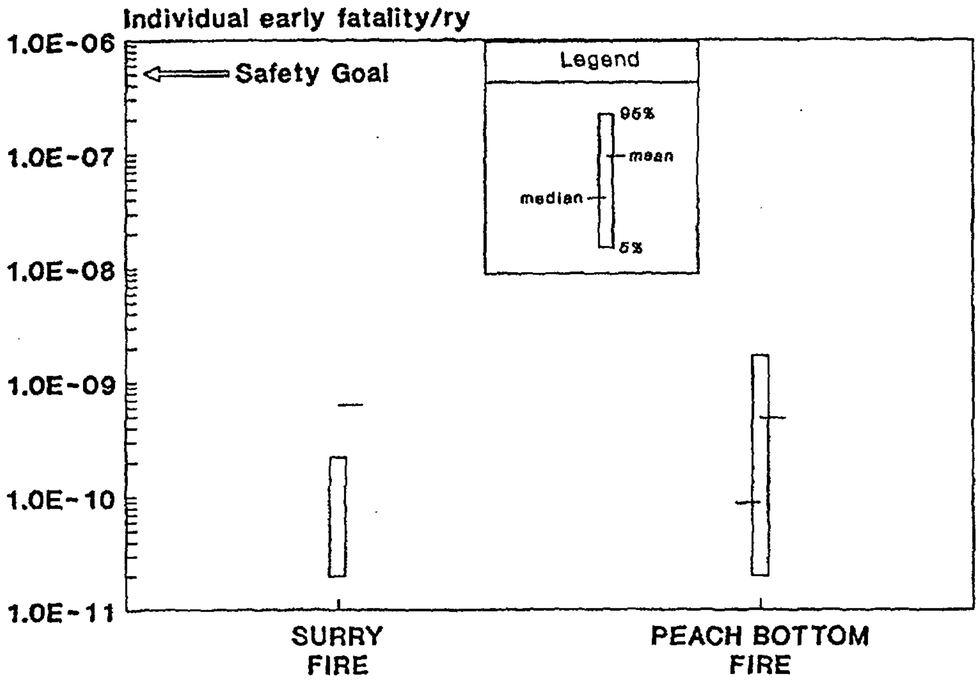
"+" indicates recalculated mean value based on recent modifications to the Zion plant (as discussed in Section C.15).

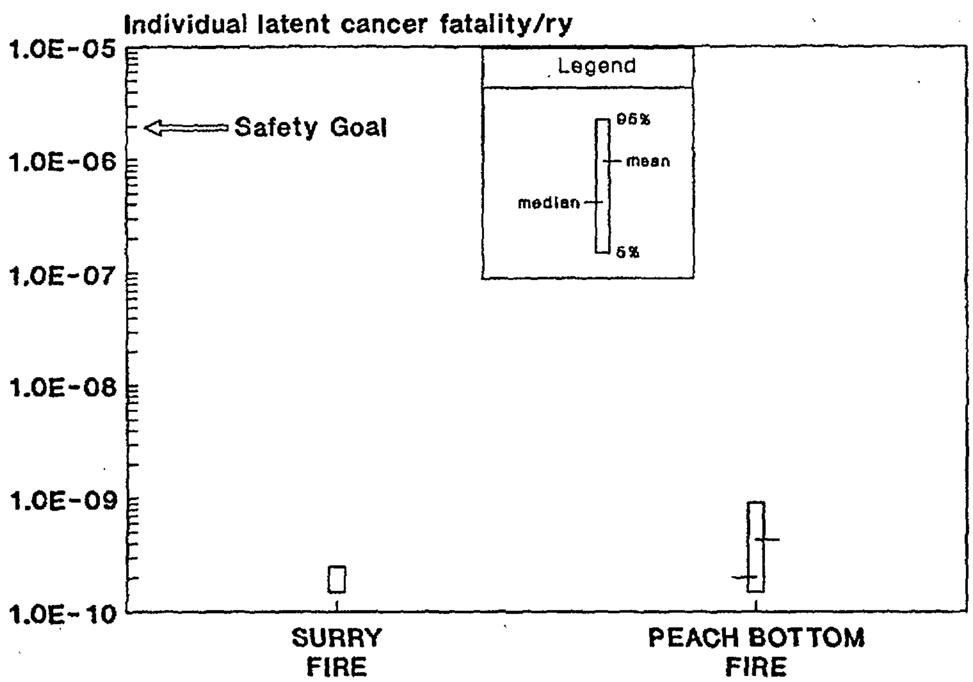
Figure 12.2 Comparison of risk results at all plants with safety goals (internal events).



Note: As discussed in Reference 12.9, estimated risks at or below 1E-7 should be viewed with caution because of the potential impact of events not studied in the risk analyses.

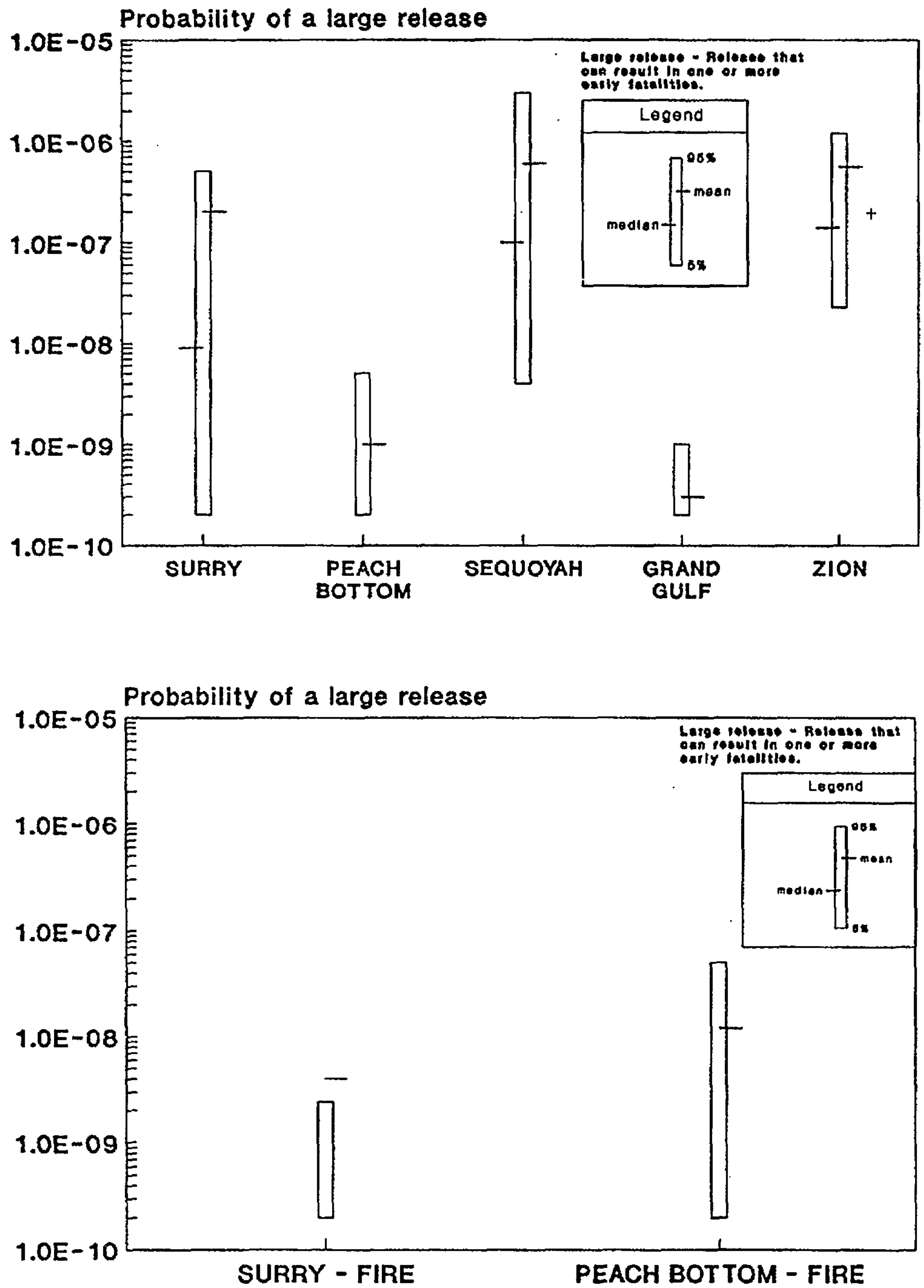
Figure 12.3 Comparison of early and latent cancer fatality risks at Surry and Peach Bottom (fire-initiated accidents).





Note: As discussed in Reference 12.9, estimated risks at or below 1E–7 should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 12.4 Comparison of risk results at Surry and Peach Bottom with safety goals (fire-initiated accidents).



Notes: As discussed in Reference 12.9, estimated risks at or below 1E-7 should be viewed with caution because of the potential impact of events not studied in the risk analyses.

"+" indicates recalculated mean value based on recent modifications to the Zion plant (as discussed in Section C.15).

Figure 12.5 Frequency of one or more early fatalities at all plants.

- The principal challenges to containment structures vary considerably among the five plants studied. Hydrogen combustion is a significant threat to the Sequoyah and Grand Gulf plants (in part because of the inoperability of ignition systems in some key accident sequences), while direct attack of the containment structure by molten core material is most important in the Peach Bottom plant. Few physical processes were identified that could seriously challenge the Surry and Zion containments.

- Emergency response parameters (warning time, evacuation speed, etc.) appear to have a significant impact on early fatality risk but almost no effect on latent cancer fatality risk.

12.3 Comparison with Reactor Safety Study

Results of the present study (for internal initiators) are compared with the Surry and Peach Bottom results in the Reactor Safety Study (RSS) in Figure 12.1. In general, for the early fatality risk measure, the Surry risk estimates in this study are lower than the corresponding RSS PWR values. Similarly, the present Peach Bottom risk estimates are lower than the RSS BWR estimates. For the latent cancer fatality risk measure, the patterns in the results are less clear; the RSS risk estimates for both of the plants lie in the upper portion of the risk estimates of this study.

Focusing on the major contributors to risk, it may be seen that, in the RSS, the Surry risk was dominated by interfacing-system LOCA (the V sequence), station blackout (TMLB'), and small LOCA sequences, with hydrogen burning and overpressure failures of containment. While the estimated risks of the interfacing-system LOCA accident sequence are lower in the present study because of a lower estimated frequency, it is still an important contributor to risk. Also important (because of their large source terms) are containment bypass accidents initiated by steam generator tube rupture, compounded by operator errors (which result in core damage) and subsequent stuck-open safety-relief valves on the secondary side. Early overpressurization containment failure at Surry is much less probable.

In the Peach Bottom analysis of the RSS, risk was dominated by transient-initiated events with loss of heat removal (TW type of sequence) and ATWS accidents with failure of containment prior to vessel breach. Dominant containment failure modes were from steam overpressurization. In the present study, risk is dominated by long-term station blackout and ATWS accident sequences. The dominant containment failure mode is drywell meltthrough.

The RSS did not perform an analysis of accidents initiated by fires. As such, comparisons of the present study's fire risk estimates with the RSS are not possible.

Since the publication of the RSS in 1975, a vast amount of work has been done in all areas of risk analysis, funded by government agencies and the nuclear industry. Major improvements have been made in the understanding of severe accident phenomenology and approaches to quantification of risk, many of which have been used in this study. These efforts have helped in lowering the estimates of overall risk levels in the present study to some extent by reducing the use of conservative and bounding types of analyses. Equally important, some plants have made modifications to plant systems or procedures based on PRAs, lessons learned from the Three Mile Island accident, etc., thus reducing risk. On the other hand, new issues have been raised and the possibility of new phenomena such as direct containment heating and drywell meltthrough has been introduced, which added to the previous estimates of risk. For issues that are not well understood, expert judgments were elicited that frequently showed diverse conclusions. The net effect of this improved understanding is that total plant risk estimates are lower than the RSS estimates, but the distributions of these risk measures are very broad.

12.4 Perspectives

As discussed above, plant-specific features contribute largely to the estimates of risks. In order to compare the variables and characteristics of the three PWR plants (Surry, Sequoyah, and Zion) and two BWR plants (Peach Bottom and Grand Gulf) in this study, the dominant contributors to early and latent cancer fatality risks for the PWRs and BWRs from internally initiated events are shown in Figures 12.6 through 12.10. Dominant contributors to risk from fire-initiated accidents for Surry and Peach Bottom are compared in Figure 12.9. Perspectives on risks for the five plants from these comparisons, supplemented by information in the supporting contractor reports (Refs. 12.1 through 12.7) are discussed below.

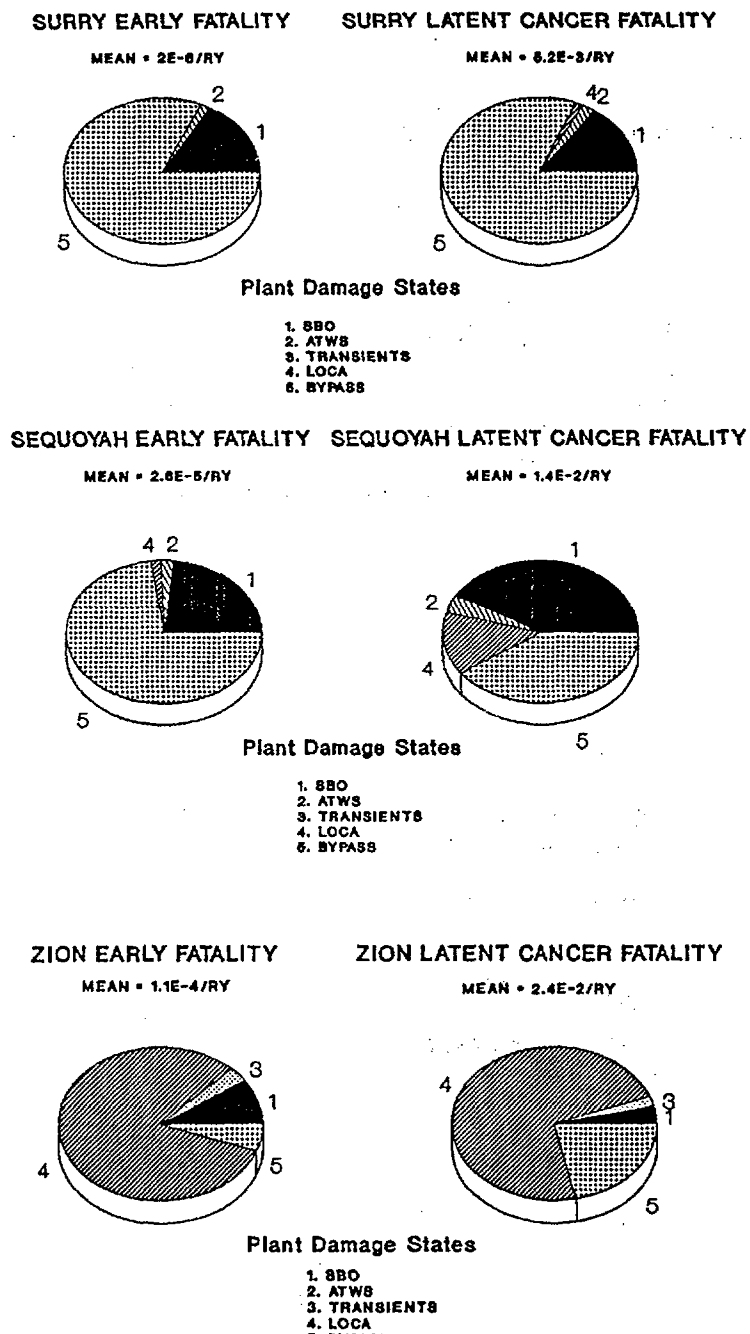


Figure 12.6 Contributions of plant damage states to mean early and latent cancer fatality risks for Surry, Sequoyah, and Zion (internal events).

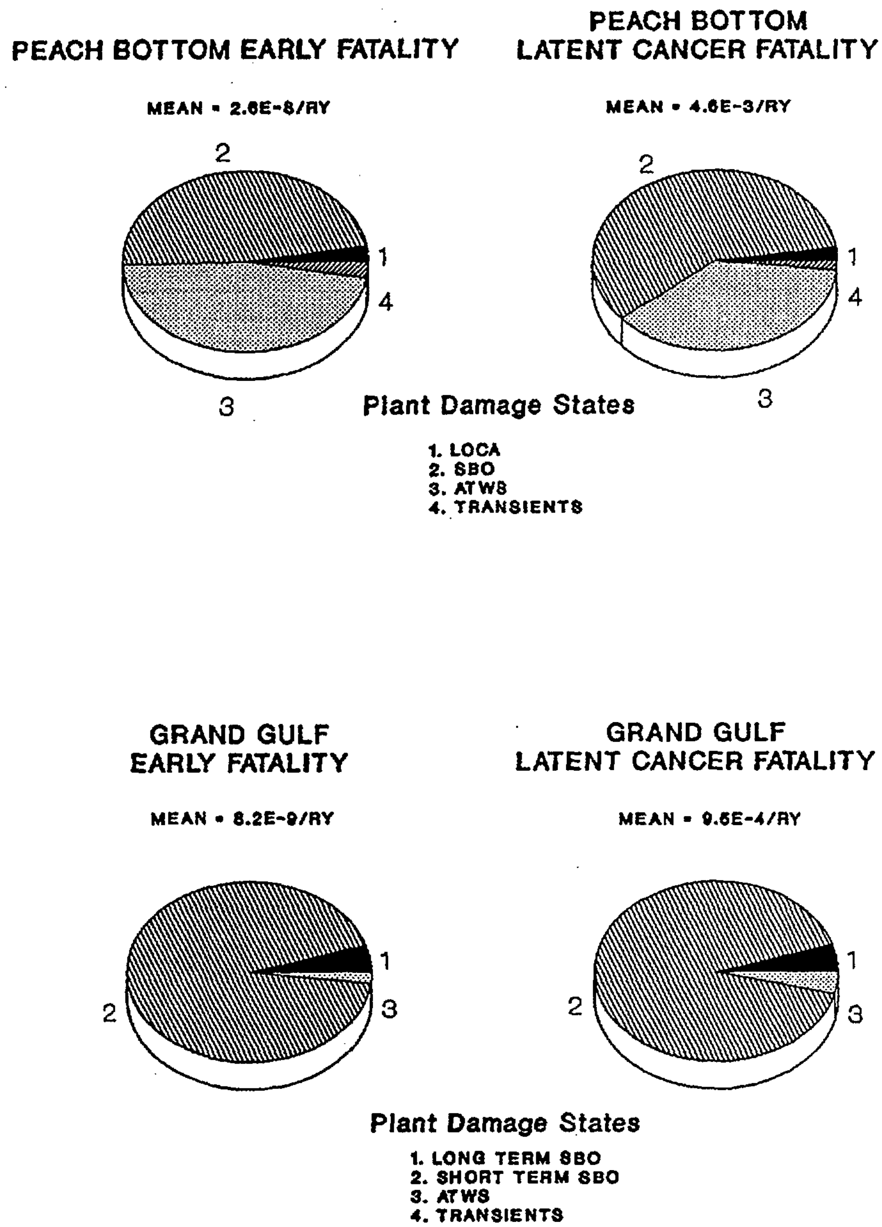


Figure 12.7 Contributions of plant damage states to mean early and latent cancer fatality risks for Peach Bottom and Grand Gulf (internal events).

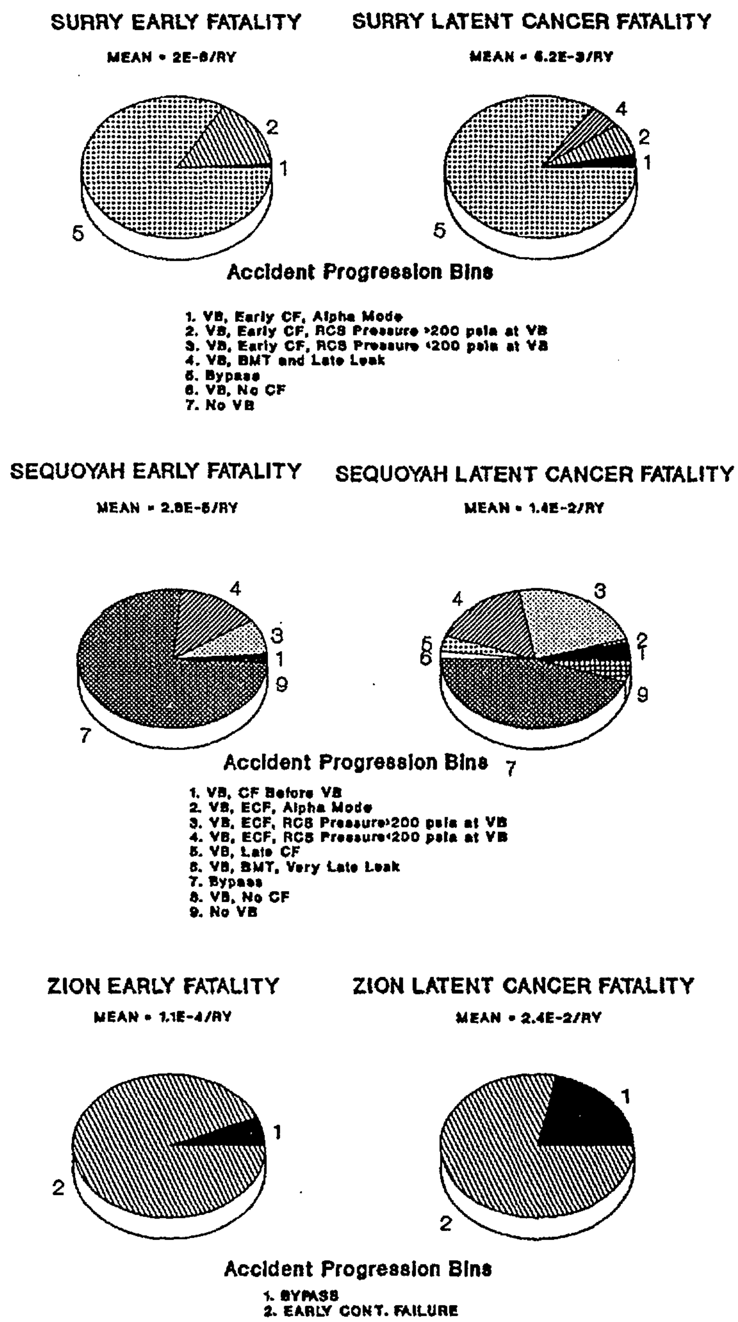


Figure 12.8 Contributions of accident progression bins to mean early and latent cancer fatality risks for Surry, Sequoyah, and Zion (internal events).

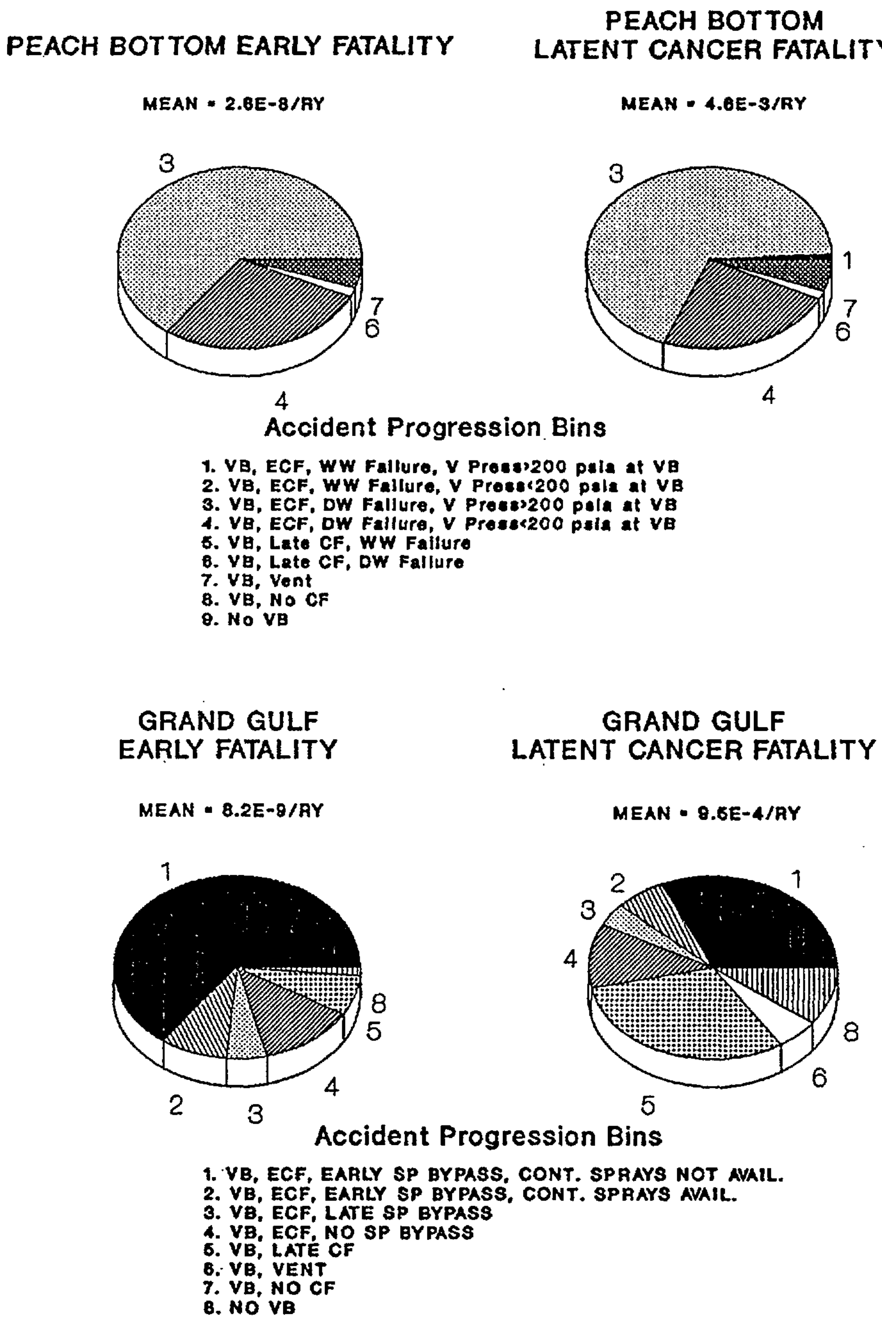


Figure 12.9 Contributions of accident progression bins to mean early and latent cancer fatality risks for Peach Bottom and Grand Gulf (internal events).

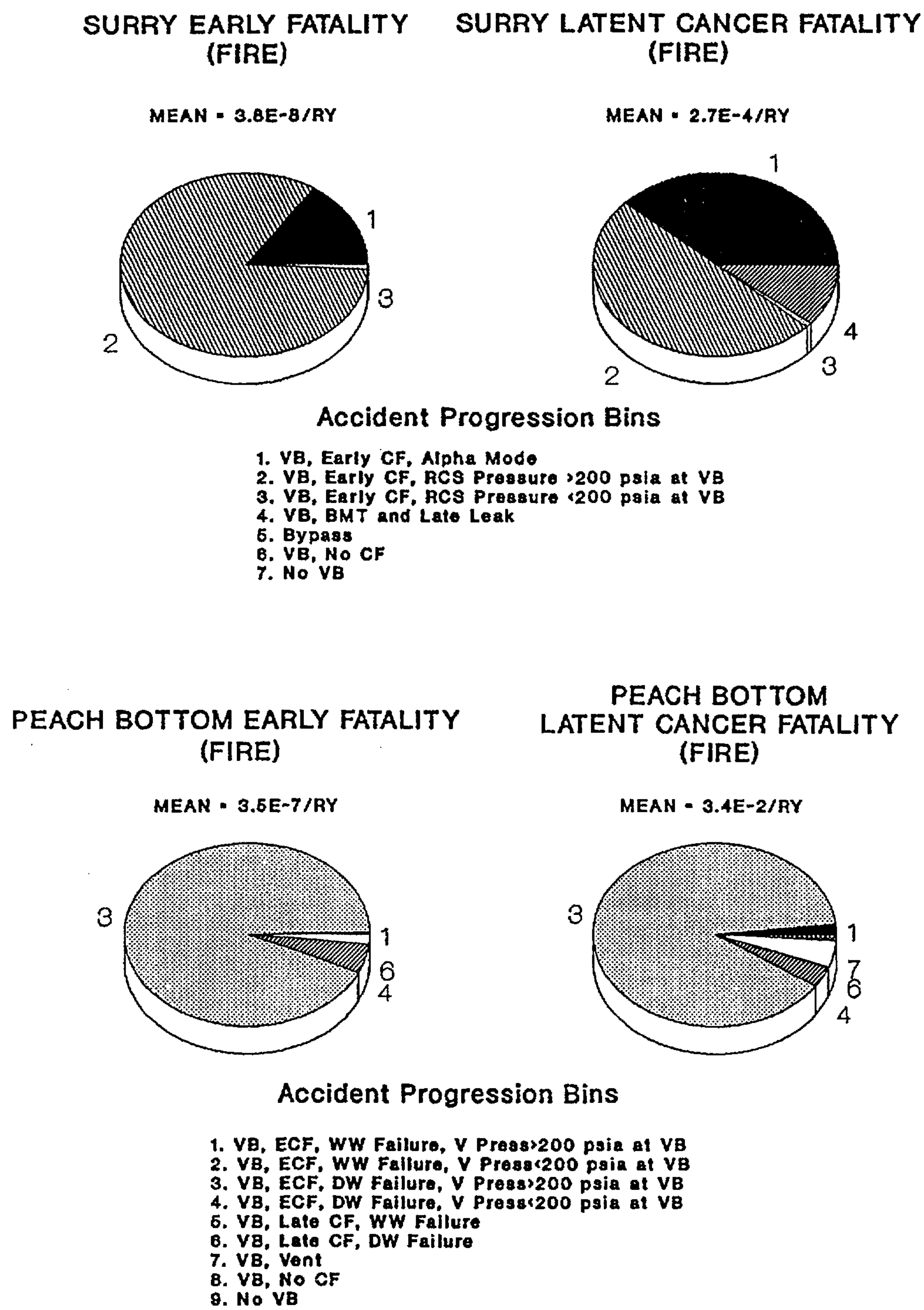


Figure 12.10 Contributions of accident progression bins to mean early and latent cancer fatality risks for Surry and Peach Bottom (fire-initiated accidents).

Accident Sequences Important to Risk

- Mean early fatality risks at Surry and Sequoyah and latent cancer fatality risk at Surry are dominated by bypass accidents (Event V and steam generator tube rupture accidents). Sequoyah latent cancer risk is dominated equally by loss of offsite power sequences and bypass accidents. The risk at Zion is dominated by medium LOCA sequences resulting from the failure of reactor coolant pump seals, induced by failures of the component cooling water system (CCWS) or service water system. Zion has the feature that CCWS (supported by the service water system) cools both the reactor coolant pump seals and high-pressure injection pump oil coolers, thus creating the potential for a common-mode failure. (As discussed in Chapter 7, steps have been taken by the plant licensee to address this dependency.)

- BWR risks are driven by events that fail a multitude of systems (i.e., reduce the redundancy through some common-mode or support system failure) or events that require a small number of systems to fail in order to get to core damage, such as ATWS sequences. The accidents important to both early fatality and latent cancer fatality risk at Peach Bottom are station blackouts and ATWS; the accident most important at Grand Gulf is station blackout.

- For the Peach Bottom plant, the estimated risks from accidents initiated by fires, while low, are greater than those from accidents initiated by internal events. Fire-initiated accidents are similar to station blackout accidents in terms of systems failed and accident progression. As such, the conditional probability of early containment failure is relatively high, principally due to the drywell shell melt-through failure mode (see Chapter 9 for additional discussion) (the conditional probability is somewhat higher because of the lower probability of ac power recovery). For the Surry plant, the fire risks are estimated to be smaller than those from internal events. This is because of two reasons: the frequency of core damage from fire initiators is lower; and fire-initiated accidents result in low conditional probabilities of early containment failure. As noted above, the internal-event risks are dominated by containment bypass accidents.

Containment Failure Issues Important to Risk

- At Surry, containment bypass events (interfacing-system LOCAs and steam generator tube ruptures) are assessed to be most important to risk. Other containment failure modes of less importance are: static failure at the containment spring line from loads at vessel breach (i.e., direct containment heating loads, hydrogen burns, ex-vessel steam explosion loads, and steam blowdown loads); or containment failure from in-vessel steam explosions (the "alpha-mode" failure of the Reactor Safety Study). These failure modes have only a small probability of resulting in early containment failure.

- At Zion, the conditional probability of early containment failure is small, comparable to that of Surry. Those containment failure modes that contribute to this small failure probability include alpha-mode failure, containment isolation failure, and overpressurization failure at vessel breach.

- In previous studies, the potential impact of direct containment heating loads was found to be very important to risk. In this study, the potential impact is less significant for the Surry and Zion plants. Reasons for this reduced importance include:

  - Temperature-induced and other depressurization mechanisms that reduce the probability of reactor vessel breach at high reactor coolant system pressure, either eliminating direct containment heating (DCH) or reducing the pressure rise at vessel breach. These depressurization mechanisms are stuck-open power-operated relief valves, reactor coolant pump seal failures, accident-induced hot leg and surge line failures, and deliberate opening of PORVs by operators; and

  - The size and the strength of the Surry containment (the maximum DCH load has only a conditional probability of 0.3 of failing the containment).

  Additional discussion of the issue of direct containment heating may be found in Section 9.4.3 and Section C.5 of Appendix C.

- At Sequoyah, containment bypass events are assessed to be most important to mean early fatality risk. Another failure important to early fatality risk is early failure of containment. In particular, the catastrophic rupture failure mode dominates early containment failures, which occur as a result of pre-vessel-breach hydrogen events and failures at vessel breach. The failures at vessel breach are the result of a variety of load sources (individually or in some combinations), including direct containment heating loads, hydrogen burns, direct contact of molten debris with the steel containment, alpha-mode failures, or loads from ex-vessel steam explosions. The bypass mode of containment failure and early containment failures dominate the mean latent cancer risk at Sequoyah and contribute about equally to this consequence measure.

- At Peach Bottom, drywell meltthrough is the most important mode of containment failure. Other containment failure modes of importance are: drywell overpressure failure, static failure of the wetwell (above as well as below the level of the suppression pool), and static failure at the drywell head.

- At Grand Gulf, the risk is most affected by containment failures in which both the drywell and the containment fail. As discussed in Chapter 9, roughly one-half the containment failures analyzed in this study also resulted in drywell failure. The principal causes of the combined failures were hydrogen combustion in the containment atmosphere and loads at reactor vessel breach (direct containment heating, ex-vessel steam explosions, or steam blowdown from the reactor vessel).

Source Term and Offsite Consequence Issues Important to Risk

- BWR suppression pools provide a significant benefit in severe accidents in that they effectively trap radioactive material (such as iodine and cesium) released early in the accident (before vessel breach) and, for some containment failure locations, after vessel breach as well.

- Accidents that bypass the containment structure compromise the many mitigative features of these structures and thus can have significant estimated radioactive releases. As noted above, such accidents dominated the risk for the Surry and Sequoyah plants.

- The design of the reactor cavity can significantly influence long-term releases of radioactive material; if large amounts of water can enter the cavity (e.g., as at Sequoyah), releases during core-concrete interactions can be significantly mitigated.

- Site parameters such as population density and evacuation speeds can have a significant effect on some risk measures (e.g., early fatality risk). Other risk measures, such as latent cancer fatality risk and individual early fatality risk, are less sensitive to such parameters. Latent cancer fatality risks are sensitive to the assumed level of interdiction of land and crops. (These issues are discussed in more detail below.)

Factors Important to Uncertainty in Risk

In order to identify the principal sources of uncertainties in the estimated risk, regression analyses have been performed for each of the plants in this study. A stepwise linear model is used, and, in general, the dependent variable is a risk measure (e.g., early fatalities per year) although some study has been done on the Surry plant using frequencies of radionuclide releases (discussed in Section 10.4.3). The independent variables consisted of individual parameters and groups of correlated parameters. Also, the analyses are generally performed for the complete risk model, although in some cases analyses are performed on specific plant damage states. The extent to which this model accounted for the overall uncertainty (the R-square value) varied considerably, from roughly 30 percent in the analysis of latent cancer fatality risk in the Sequoyah plant to roughly 75 percent in the analysis of early fatality risk in the Surry plant.

The results of the regression analyses indicate the following:

- For Surry, the uncertainty in all risk measures is dominated by the uncertainties in parameters determining the frequencies of containment bypass accidents (interfacing-system LOCA and steam generator tube rupture (SGTR)) and the radioactive release magnitudes of these accidents. More specifically, the most important parameters are the initiating event frequencies for these bypass accidents, the fraction of the core radionuclide inventory released into the vessel, and the fraction of material in the vessel in an SGTR-initiated core damage accident that is released to the environment. With the high risk importance of bypass accidents, it is not surprising that uncertainties in bypass accident parameters are important to risk uncertainty, while other parameters such as those relating to source terms in containment, containment strength, etc., are not found to be important.

- For Zion, the regression analyses also indicated that accident frequency and source term parameter uncertainties were most important. More specifically, the most important parameters were the initiating event frequencies for loss of component cooling water (CCW)/service water (SW), the failure to recover CCW/SW, the fraction of the core radionuclide inventory released into the vessel, the radionuclide containment transport fraction at vessel breach, and the fraction of radionuclides released to the environment through the steam generators. The importance of the loss of CCW/SW frequencies is not surprising, given the large contribution of accidents initiated by these events to the core damage frequency. Also, those source term parameters that influence the release fractions for early containment failure and bypass events are not surprisingly important to some risk measures. The only accident progression parameter that was demonstrated to be important to the uncertainty in risk was the probability of vessel and containment breach by an in-vessel steam explosion. This result occurs because the probability of early containment failure from all other causes is extremely low at Zion, so that (at these very low probability levels) uncertainty in the in-vessel steam explosion failure mode becomes more significant. The importance of the steam explosion failure mode is also more significant because the accident progression analysis for Zion indicates that the reactor coolant system (RCS) is not likely to be at high pressure when vessel breach occurs. This means that loads at vessel breach from direct containment heating are likely to be smaller than would have been the case if RCS pressure were high. Also, at low RCS pressure, the probability of triggering an in-vessel steam explosion is increased.

- For Sequoyah, the regression analysis for the complete risk model did not account for a large fraction of the uncertainty. As such, regression analyses were performed for individual plant damage states (PDSs). For the containment bypass PDSs (which dominated the mean risk at Sequoyah), the most important uncertainties related to accident frequency and source term issues. More specifically, for the interfacing-system LOCA PDS, the most important parameter uncertainties were those for the initiating event frequency, the probability that releases will be scrubbed by fire sprays in the vicinity of the break, and the decontamination factor of the fire sprays. For the SGTR-initiated core damage accident, the most important parameters are the initiating event frequency, the fraction of the core radionuclide inventory released into the vessel, and the fraction of material in the vessel that is released to the environment.

  For the station blackout, LOCA, and transient plant damage states, the uncertainty in early fatality risk is accounted for by parameters from the accident frequency, accident progression, and source term analysis, with none of these groups or any small set of parameters dominating. In this circumstance, the parameters relating to the containment failure pressure, the fraction of the core participating in a high-pressure melt ejection, and the pressure rise at vessel breach for low-pressure accident sequences appeared as somewhat important for each of these plant damage states (but, again, did not by themselves or in combination dominate the uncertainty estimation).

- For Peach Bottom, the regression analysis for the complete internal-event model indicated that the risk uncertainty is dominated by uncertainties in radioactive release uncertainties—more specifically, the dominating parameters relating to the fraction of the core radionuclide inventory released into the vessel before vessel breach, the fraction of the radionuclide inventory released during core-concrete interaction that is released from containment, and the fraction of the radionuclide inventory remaining in the core material at the initiation of core-concrete interaction that is released during that interaction.

  The regression analysis on the fire risk model does not show such a clear domination by any parameters. The early fatality risk uncertainty is dominated by radioactive release parameters (the fraction of core radionuclide inventory released to the vessel before vessel breach, the fraction of radionuclide inventory remaining in the core material at the initiation of core-concrete interaction that is released during that interaction, and the fraction of the radionuclide inventory released during core-concrete interaction that is released from containment). The latent cancer fatality risk uncertainty is dominated by accident frequency parameters (fire initiating event frequencies, diesel generator failure-to-run probability).

- For Grand Gulf, the uncertainty in early health effect parameters (early fatalities and individual early fatalities within 1 mile) is not dominated by any small set of parameters. Rather, it is accounted for by a number of parameters that determine the frequencies and radioactive release magnitudes of those events leading to early containment failure, such as the amount of hydrogen generated during the in-vessel portion of the accident progression, and the frequency of loss of offsite power. The uncertainties in the other risk measures are dominated by uncertainties in accident frequency parameters (including loss of offsite power frequency, diesel generator failure-to-start probability, diesel generator failure-to-run probability, and the probability that the batteries fail to deliver power when needed).

Impact of Emergency Response and Protective Action Guide Options

Sensitivity calculations were performed as a part of this study to assess the impacts of different emergency response and protective action guide options on estimates of risks for the five plants.

Emergency Response Options

In order to study the effects of emergency response options under severe accident conditions on public risk, the plants were analyzed using the following assumptions, and changes in the early fatality risk were calculated:

- Base Case: 99.5 percent evacuation from 0 to 10 miles
- Option 1: 100 percent evacuation from 0 to 10 miles
- Option 2: 0 percent evacuation with early relocation from high contamination areas
- Option 3: 100 percent sheltering
- Option 4: 100 percent evacuation from 0 to 5 miles and 100 percent sheltering from 5 to 10 miles

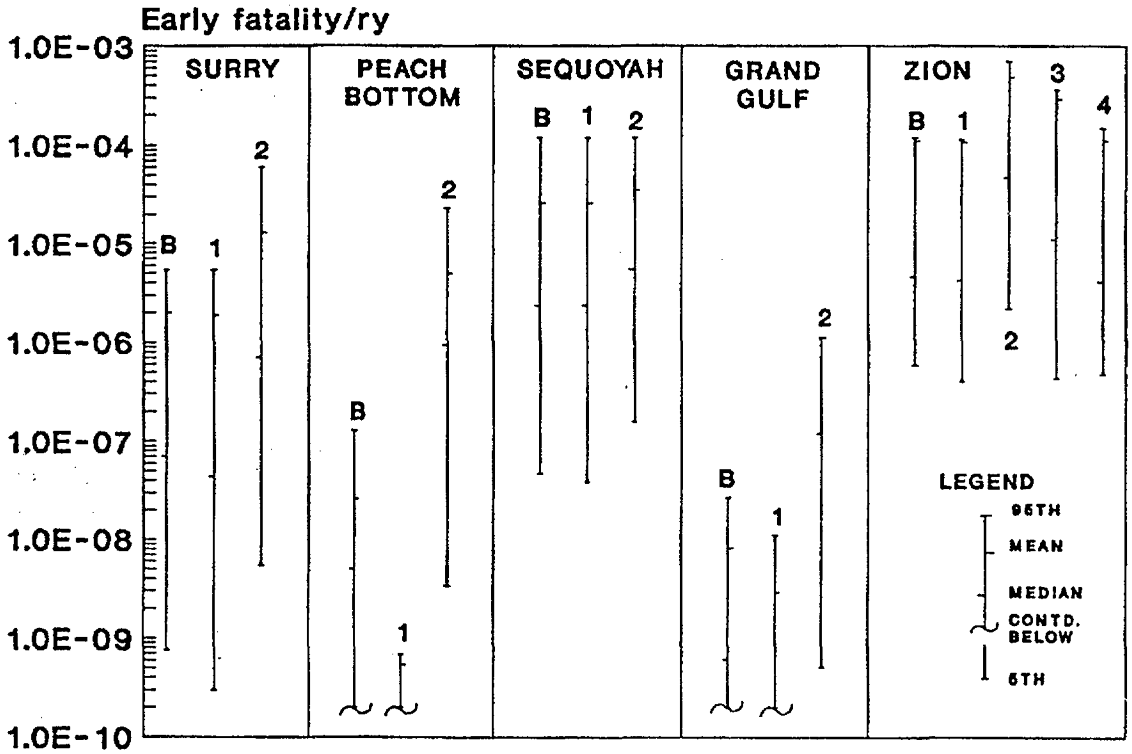
The last two options are used in the Zion plant analysis only. Results of the analyses are presented in Figure 12.11.

As discussed in Section 11.3, radionuclide release magnitudes associated with the early phase of an accident for Peach Bottom and Grand Gulf are typically smaller than those for the other three plants because of the mitigative effects of suppression pool scrubbing. The source term groups for Peach Bottom and Grand Gulf were typically found to have longer warning times than for the PWRs studied because the accident sequences developed more slowly. Further, Peach Bottom and Grand Gulf have very low surrounding population densities, which leads to shorter evacuation delays and higher evacuation speeds. The effect of all these considerations is that, for Peach Bottom and Grand Gulf, evacuation is more effective in reducing early fatality risk than for Surry, Sequoyah, and Zion.

For Surry and Sequoyah, the risk-dominant accident is the interfacing-system LOCA (the V sequence). This accident has a very short warning time, and, consequently, evacuation actions are not very effective. Also for Sequoyah, some high-consequence releases occur from containment failure at vessel breach; these releases are highly energetic and cause plume rise. This reduces early fatality risk, as is indicated in the case of Option 2 for Sequoyah; however, this also reduces the effectiveness of evacuation. Further details on emergency response options are provided in Chapter 11.

Protective Action Options

In this study an interdiction criterion of 4 rems (effective dose equivalent (EDE)) in 5 years has been used for groundshine and inhalation of resuspended radionuclides. Sensitivity calculations have been performed using the equivalent of the Reactor Safety Study (RSS) criterion, i.e., 25-rem EDE in 30 years. The impact of such an alternative criterion on mean latent cancer fatality risk is shown in Figure 12.12. As may be seen, the RSS criterion is less restrictive than the criterion used in this study, and the corresponding latent cancer fatalities using the RSS criterion are higher by 12 percent (for Grand Gulf) to 47 percent (for Peach Bottom).



BASE CASE (B)

99.5% Evacuation from 0 to 10 miles

EMERGENCY RESPONSE OPTIONS (1 TO 4)

1. 100% Evacuation from 0 to 10 miles
2. 0% Evacuation with early relocation from high contamination areas
3. 100% Sheltering
4. 100% Evacuation from 0 to 5 miles, and 100% sheltering from 5 to 10 miles

Note: As discussed in Reference 12.9, estimated risks at or below 1E–7 should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 12.11 Effects of emergency response assumptions on early fatality risks at all plants (internal events).

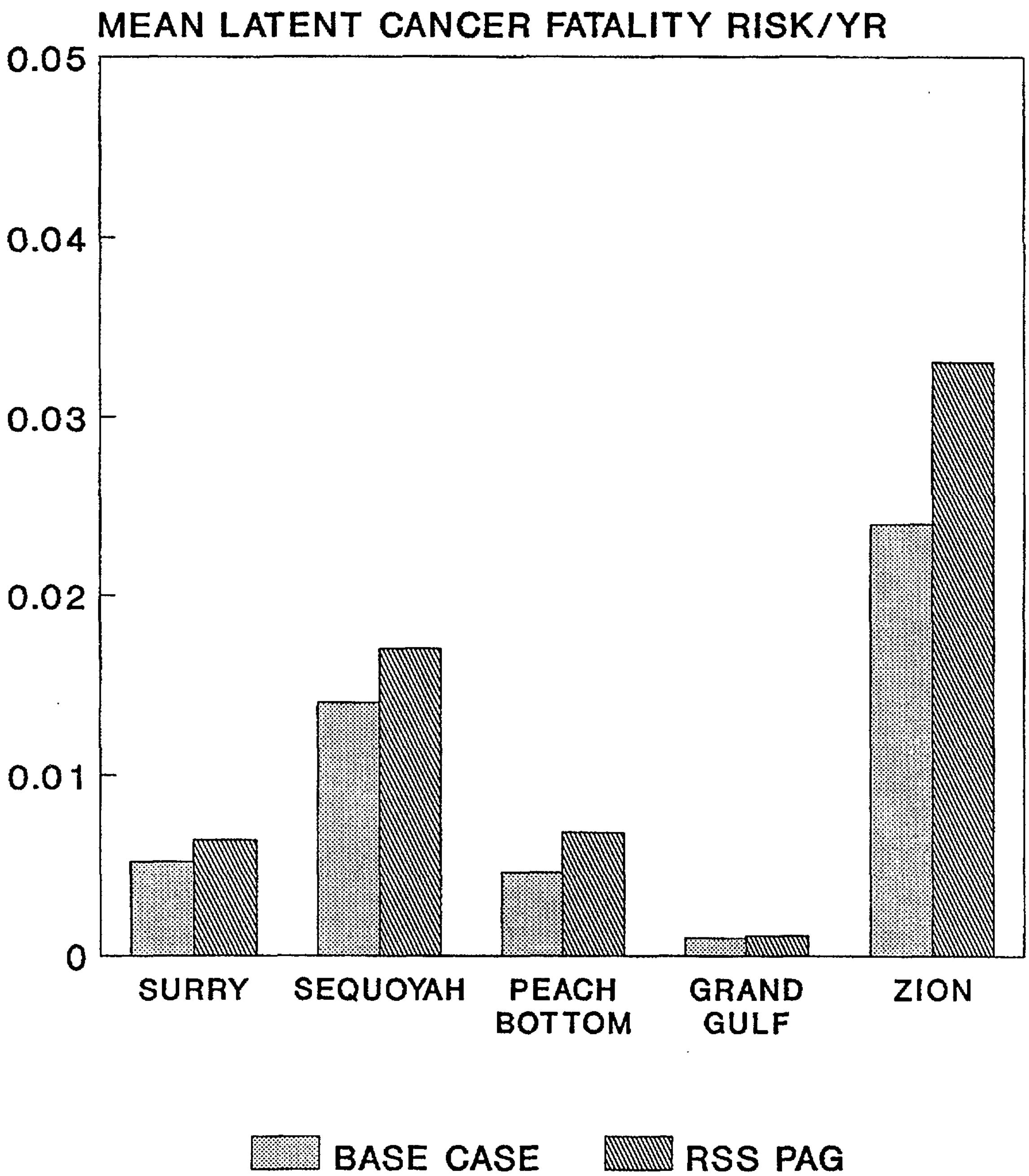


Figure 12.12 Effects of protective action assumptions on mean latent cancer fatality risks at all plants (internal events).

REFERENCES FOR CHAPTER 12

12.1 E. D. Gorham-Bergeron et al., "Evaluation of Severe Accident Risks: Methodology for the Accident Progression, Source Term, Consequence, Risk Integration, and Uncertainty Analyses," Sandia National Laboratories, NUREG/CR-4551, Vol. 1, Draft Revision 1, SAND86-1309, to be published.*

12.2 F. T. Harper et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," Sandia National Laboratories, NUREG/CR-4551, Vol. 2, Revision 1, SAND86-1309, December 1990.

12.3 R. J. Breeding et al., "Evaluation of Severe Accident Risks: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 3, Revision 1, SAND86-1309, October 1990.

12.4 A. C. Payne, Jr., et al., "Evaluation of Severe Accident Risks: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4551, Vol. 4, Draft Revision 1, SAND86-1309, to be published.*

12.5 J. J. Gregory et al., "Evaluation of Severe Accident Risks: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 5, Revision 1, SAND86-1309, December 1990.

12.6 T. D. Brown et al., "Evaluation of Severe Accident Risks: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 6, Draft Revision 1, SAND86-1309, to be published. *

12.7 C. K. Park et al., "Evaluation of Severe Accident Risks: Zion Unit 1," Brookhaven National Laboratory, NUREG/CR-4551, Vol. 7, Draft Revision 1, BNL-NUREG-52029, to be published.*

12.8 USNRC, "Safety Goals for the Operation of Nuclear Power Plants; Policy Statement," *Federal Register*, Vol. 51, p. 30028, August 21, 1986.

12.9 H. J. C. Kouts et al., "Special Committee Review of the Nuclear Regulatory Commission's Severe Accident Risks Report (NUREG-1150)," NUREG-1420, August 1990.

12.10 USNRC, "Reactor Safety Study—An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

*Available in the NRC Public Document Room, 2120 L Street NW., Washington, DC.

13. NUREG–1150 AS A RESOURCE DOCUMENT

13.1 Introduction

NUREG–1150 is one element of the NRC's program to address severe accident issues. The entire program was discussed in a staff document entitled "Integration Plan for Closure of Severe Accident Issues" (SECY–88–147) (Ref. 13.1). NUREG–1150 is used to provide a snapshot of the state of the art of probabilistic risk analysis (PRA) technology, incorporating improvements since the issuance of the Reactor Safety Study (Ref. 13.2). This chapter discusses the results of NUREG–1150 (and its supporting contractor studies, Refs. 13.3 through 13.16) as a resource document and examines the extent to which information provided in the document can be applied in regulatory activities. This is accomplished by applying NUREG–1150 results and principles to selected regulatory issues to illustrate how the information and insights described in Chapters 3 through 12 of this document can be used in the regulatory process. The discussion will concentrate on technical issues although it is recognized that there are other issues (e.g., legal, procedural) that must be taken into account when making regulatory decisions.

This report includes an examination of the severe accident frequencies and risks and their associated uncertainties for five licensed nuclear power plants and uses the latest source term information available from both the NRC and its contractors and the nuclear industry. The information in the report provides a valuable resource and insights to the various elements of the severe accident integration plan. The information provided and how it will be used include the following:

- Probabilistic models of the spectrum of possible accident sequences, containment events, and offsite consequences of severe accidents for use in:
  - Development of guidance for the individual plant examinations of internally and externally initiated accidents;
  - Accident management strategies;
  - Analysis of the need and appropriate means for improving containment performance under severe accident conditions;
  - Characterization of the importance of plant operational features and areas potentially requiring improvement;
  - Analysis of alternative safety goal implementation strategies; and
  - Emergency preparedness and consequences.
- Data on the major contributing factors to risk and the uncertainty in risk for use in:
  - Prioritization of research;
  - Prioritization of generic issues; and
  - Use of PRA in inspection.

In the following sections, these uses will be discussed in greater detail, using examples based on the risk analysis results discussed in previous chapters.

13.2 Probabilistic Models of Accident Sequences

NUREG–1150 identifies the dominant accident sequences and plant features contributing significantly to risk at a given plant as well as the plant models used in the study. The plant models and results underlying the report can be used to support the development of staff guidance on licensee-performed studies (individual plant examinations, accident management studies) and staff work in other areas related to severe accidents (e.g., improving containment performance under severe accident conditions). Such uses are discussed in greater detail in the following sections.

13.2.1 Guidance for Individual Plant Examinations

Plant-specific PRAs have yielded valuable perspectives on unique plant vulnerabilities. The NRC and the nuclear industry both have considerable experience with plant-specific PRAs. This experience, coupled with the interactions of NRC and the nuclear industry on severe accident issues, have resulted in the Commission's formulating an integrated systematic approach to an examination of each nuclear power plant now operating or under construction for possible significant risk contributions (sometimes called "outliers") that might be plant specific and might be missed without a systematic approach. In November 1988, the NRC requested (by generic letter) that each licensed nuclear power plant perform an individual plant examination (IPE) to identify any plant-specific vulnerabilities to severe accidents (Ref. 13.17). The technical data generated in the course of preparing NUREG–1150 on severe accident frequencies, risks, and important uncertainties were used in developing the analysis requirements described in the IPE generic letter and the supplemental guidance on the IPE external-event analysis (Ref. 13.18).* These studies will also aid the staff in evaluating individual submittals, assessing the adequacy of the identification of plant-specific vulnerabilities by the licensee, and evaluating any associated potential plant modifications.

The extent to which NUREG–1150 results are applicable to different classes of reactors or to operating U.S. light-water reactors as a group is illustrated in Table 13.1. The generic insights presented in NUREG–1150 are indicative of items that may be applicable within a class of plants. This includes the identification of possible vulnerabilities that may exist in plants of similar design. These insights cannot be assumed to apply to a given plant without consideration of plant design and operational practices because of the design differences that exist in U.S. plants, particularly those involving ancillary support systems (e.g., ac power, component cooling water) for the engineered safety features and differences in details of containment design.

For some issues, the state of knowledge is very limited, and it is not possible to identify plant-specific features that may influence the issue because sensitivity analyses have not been performed. In other cases, the methodology is broadly applicable, but the results are highly plant specific. In spite of the plant-specific nature of many of the results, much can be learned from one plant that can be applied to another. Example types of generic applicability are presented in Table 13.1.

The NUREG–1150 methods refer not only to the analytical techniques employed but the general structure and framework upon which the analyses were conducted. These methods include the uncertainty analysis, expert elicitation methods, accident progression event tree analysis, and source term modeling. The general approaches adopted in these analysis procedures are not plant specific and are therefore adaptable to other plant analyses.

*In addition, NUREG–1150 provides extensive and detailed analyses of five nuclear power plants and thus offers licensees of those plants an opportunity to use these studies in developing their IPEs and submitting them on an expedited basis.

As noted above, plant-specific PRAs have yielded valuable perspectives on unique plant vulnerabilities. These perspectives are, in general, not directly applicable to other plants, although they provide useful information to the study of plants of similar NSSS (nuclear steam supply system) and containment design. At the present time, the principal contributors to the likelihood of a core damage accident at boiling water reactors (BWRs) include sequences related to station blackout or anticipated transients without scram (ATWS). Accident sequences making important contributions to the frequency of core damage accidents at pressurized water reactors (PWRs) include those initiated by a variety of electrical power system disturbances (loss of a single ac bus, which initiates a transient; loss of offsite portions of the equipment needed to respond to the transient; loss of offsite power; and complete station blackout), small loss-of-coolant accidents (LOCAs), loss of coolant support systems such as the component cooling water system, ATWS, and interfacing-system LOCAs or steam generator tube ruptures in which reactor coolant is released outside the containment boundary. All have the potential for being important at PWRs.

NUREG–1150 provides a wide spectrum of phenomenological and operational data (much of it of a very detailed nature). For example, information on hydrogen generation has been compiled from experimental and calculational results as well as interpretations of these data by experts. This data base provides an important source of information that may be used for NSSS containment types similar to those studied here but is somewhat less applicable for different NSSS containment types. The operational data base includes component failure rates, maintenance times, and initiating-event frequency data. Much of these data are generic in nature and thus applicable for selected classes of plants.

The analyses presented in Chapters 3 through 7, when combined with the information gained from earlier PRA work sponsored by both NRC (e.g., Ref. 13.19) and utilities, make it clear that the quantitative results (core damage frequencies and risk results) calculated for internal and external initiators cannot be considered applicable to another plant, even if the plant has a similar NSSS design and the same architect-engineer was involved in the design of the balance of plant.

Table 13.1 Utility of NUREG-1150 PRA process to other plant studies.

Example Results	Applicability	
	Class of Plants	Plant Population
1. Methods (e.g., uncertainty, elicitation, event tree/fault tree)	high	high
2. General perspectives (e.g., principal contributors to core damage frequency and risk)	medium	low
3. Supporting data base on design features, operational characteristics, and phenomenology (e.g., hydrogen generation in core damage accidents, operational data)	high	medium
4. Quantitative results (e.g., core damage frequency, containment performance, risk)	low	low

Site-specific requirements and differing utility requirements often lead to significant differences in support system designs (e.g., ac power, dc power, service water) that can significantly influence the response of the plant to various potential accident-initiating events. Further, different operational practices, including maintenance activities and techniques for monitoring the operational reliability of components or systems can have a significant influence on the likelihood or severity of an accident.

13.2.2 Guidance for Accident Management Strategies

Certain preparatory and recovery measures can be taken by the plant operating and technical staff that could prevent or significantly mitigate the consequences of a severe accident. Broadly defined, such "accident management" includes the measures taken by the plant staff to (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) maintain containment integrity as long as possible, and finally (4) minimize the consequences of offsite releases. In addition, accident management includes certain measures taken before the occurrence of an event (e.g., improved training for severe accidents, hardware or procedure modifications) to facilitate implementation of accident management strategies. With all these factors taken together, accident management is viewed as an important means of achieving and maintaining a low risk from severe accidents.

Under the staff program, accident management programs will be developed and implemented by licensees. The NRC will focus on developing the regulatory framework under which the industry programs will be developed and implemented, as well as providing an independent assessment of licensee-proposed accident management capabilities and strategies. NUREG-1150 has been used by the NRC staff to support the development of the accident management program. NUREG-1150 methods provide a methodological framework that can be used to evaluate particular strategies, and the current results provide some insights into the efficacy of strategies in place or that might be considered at the NUREG-1150 plants. Thus, the NUREG-1150 methods and results will support a staff review of licensee accident management submittals.

PRA information has been used in the past to influence accident management strategies; however, the methods used in NUREG-1150 can bring added depth and breadth to the process, along with a detailed, explicit treatment of uncertainties. The integrated nature of the methods is particularly important, since actions taken during early parts of an accident can affect later accident progression and offsite consequences. For example, an accident management strategy at a BWR may involve opening a containment vent. This action can affect such things as the system response and core damage frequency, the retention of radioactive material within the containment, and the timing of radionuclide releases (which impacts evacuation strategies). It is possible that actions to reduce the core damage frequency can yield accident sequences of lower frequency but with much higher consequences. All these factors need to be considered in concert when developing appropriate venting strategies. The treatment of uncertainties is another key aspect of accident management. Generally, procedures are developed based on "most likely" or "expected" outcomes. For severe accidents, the outcomes are particularly uncertain. PRA models and results, such as those produced in the accident progression event trees, can identify possible alternative outcomes for important accident sequences. By making this information available to operators and response teams, unexpected events can be recognized when they occur, and a more flexible approach to severe accidents can be developed. The recent trend toward symptom-based, as opposed to event-based, procedures is consistent with this need for flexibility.

To demonstrate the potential benefits of an accident management program, some example calculations were performed, as documented in Reference 13.20. For this initial demonstration, these calculations were limited to the internal-event accident sequence portion of the analysis. Further, the numerical results presented are "point estimates" of the core damage frequency as opposed to mean frequency estimates. Selected examples from the initial analysis are presented below.

Effect of Firewater System at Grand Gulf

The first NUREG-1150 analysis of the Grand Gulf plant (Ref. 13.21) did not credit use of the firewater system for emergency coolant injection because of the unavailability of operating procedures for its use in this mode and the difficulties in physically configuring its operation. However, since that time, the licensee has made significant system and procedural modifications. As a result, the firewater system at Grand Gulf can now be used as a backup source of low-pressure coolant injection to the reactor vessel. The system would be used for long-term accident sequences, i.e., where makeup water was provided by other injection systems for several hours before their subsequent failure. The firewater system primarily aids the plant during station blackout conditions and is considered a last resort effort.

An examination has been made of the benefit of these licensee modifications to the Grand Gulf plant. As shown in Figure 13.1, these analyses showed that the total core damage frequency was reduced from 4E-6 to 2E-6 per reactor year because of these changes.

Effect of Feed and Bleed on Core Damage Frequency at Surry

The NUREG-1150 analysis for Surry includes the use of feed and bleed cooling for those sequences in which all feedwater to the steam generators is lost (thus causing their loss as heat removal systems). Feed and bleed cooling restores heat removal from the core using high-pressure injection (HPI) to inject into the reactor vessel and the power-operated relief valves (PORVs) on the pressurizer to release steam and regulate reactor coolant system pressure.

An examination has been made to determine to what extent feed and bleed cooling decreases core damage frequency at Surry. The current Surry model includes two basic events representing failure modes for feed and bleed cooling in the event of a loss of all feedwater. These modes are: operator failure to initiate high-pressure injection and operator failure to properly operate the PORVs. In order to examine the impact of feed and bleed cooling, both basic events were assumed to always occur. As shown in Figure 13.1, the resulting total core damage frequency for Surry (if feed and bleed cooling were not available) then increases by roughly a factor of 1.3. That is, the availability of the feed and bleed core cooling option in the Surry design and operation is estimated to reduce core damage frequency from 4E-5 to 3E-5 per reactor year.

Gas Turbine Generator Recovery Action at Surry

The present NUREG-1150 modeling and analysis of the Surry plant have not considered the benefits of using onsite gas turbine generators for recovery in the event of station blackout accidents. Both a 25 MW and a 16 MW gas turbine generator are available to provide emergency ac power to safety-related and non-safety-related equipment. These generators were not included in the analysis because, as currently configured, they would not be available to mitigate important accident sequences.

An examination has been made of the effect on core damage frequency at Surry of including the gas turbine generators as a means of recovery from station blackout sequences. To give credit for the addition of one generator for emergency ac power, it is assumed that Surry plant personnel have the authority to start the gas turbines when required and that 1 hour is required to start the gas turbines and energize the safety buses. In the analysis, the gas turbines were assumed to be available 90 percent of the time.

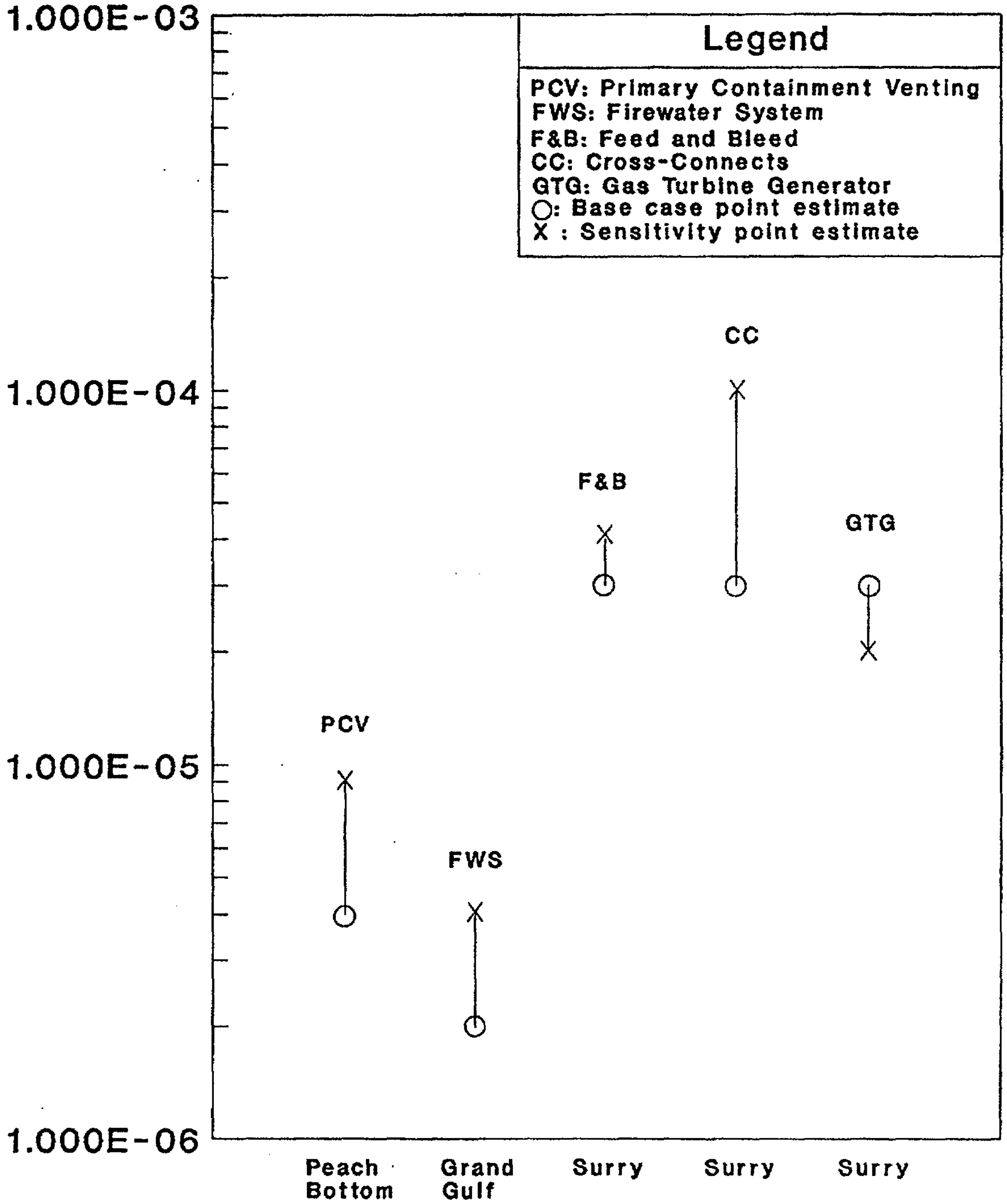


Figure 13.1 Benefits of accident management strategies.

The use of the onsite gas turbine was estimated to reduce core damage frequency from 3E-5 to 2E-5 per reactor year.

High-Pressure Injection and Auxiliary Feedwater Crossconnects at Surry

The Surry Unit 1 plant is configured to recover from loss of either the high-pressure injection (HPI) system or the auxiliary feedwater (AFW) system by operator-initiated crossconnection to the analogous system at Unit 2. While these actions provide added redundancy to these systems, new failure modes (e.g., flow diversion pathways) that were included in the modeling process for Surry have been created. The alignment of the Unit 1 and Unit 2 HPI and AFW systems for crossconnect injection is modeled as a recovery action.

Analysis of the importance of crossconnect injection at Surry includes two parts. First, credit for crossconnect injection was removed from all applicable dominant sequences, which were then requantified. Second, sequences that were previously screened out of the analysis were checked to determine if they would become dominant in the absence of crossconnect injection. As shown in Figure 13.1, the point estimate of the total core damage frequency without crossconnects is 1E-4, compared to the value of 3E-5 for internally initiated events in the base case.

Primary Containment Venting at Peach Bottom

The primary containment venting (PCV) system at Peach Bottom is used to prevent primary containment overpressurization during accident sequences in which all containment heat removal is lost. Most sequences of this type involve failure of the residual heat removal systems. Because of the existence of this venting capability, no such accident sequences appeared as dominant in the NUREG-1150 analysis for Peach Bottom.

The effect of the PCV system on the core damage frequency at Peach Bottom was determined by examining the sequences screened out in the NUREG-1150 analysis that included the PCV system as an event (primarily the sequences involving loss of containment heat removal). Credit for the PCV system was removed from these sequences, which were then summed and added to the current point estimate of the core damage frequency. As shown in Figure 13.1, this results in a point estimate of the Peach Bottom core damage frequency without containment venting of 9E-6, about a factor of 2.6 increase over the NUREG-1150 value of 4E-6.

13.2.3 Improving Containment Performance

The NRC has performed an assessment of the need to improve the capabilities of containment structures to withstand severe accidents (Ref. 13.1). Staff efforts focused initially on BWR plants with a Mark I containment, followed by the review of other containment types. This program was intended to examine potential enhanced plant and containment capabilities and procedures with regard to severe accident mitigation. NUREG-1150 provided information that served to focus attention on areas where potential containment performance improvements might be realized. NUREG-1150 as well as other recent risk studies indicate that BWR Mark I risk is dominated by station blackout and anticipated transient without scram (ATWS) accident sequences. NUREG-1150 further provided a model for and showed the benefit of a hardened vent for Peach Bottom (discussed above and displayed in Figure 13.1). The staff is currently pursuing regulatory actions to require hardened vents in all Mark I plants, using NUREG-1150 and other PRAs in the cost-benefit analysis.

The NUREG-1150 accident progression analysis models were used by the staff and its contractors in the evaluation of possible containment improvements for the PWR ice condenser and BWR Mark III designs. The result of the staff reviews of these designs (and all others except the Mark I) was that potential improvements would best be pursued as part of the individual plant examination process (discussed in Section 13.2.1).

13.2.4 Determining Important Plant Operational Features

NUREG-1150 will provide a source of information for investigating the importance of operational safety issues that may arise during day-to-day plant operations. The NUREG-1150 models, methods, and results have already been used to analyze the importance of venting of the suppression pool, the importance of keeping the PORVs and atmospheric dump valves unblocked, the importance of operational characteristics of the ice condenser containment design, the importance of operator recovery during an accident sequence, and the importance of crossties between systems. These operational and system characteristics, as well as many others, are described in detail in Chapters 3 through 7. For example, characteristics of the Surry plant design and operation that have been found to be important include crossties between units, diesel generators, reactor coolant pump seals, battery capacity, capability for feed and bleed core cooling, subatmospheric containment operation, post-accident heat removal system, and reactor cavity design.

13.2.5 Alternative Safety Goal Implementation Strategies

On August 21, 1986, the Commission published a Policy Statement on Safety Goals for the Operation of Nuclear Power Plants (Ref. 13.22). In this statement, the Commission established two qualitative safety goals supported by two risk-based quantitative objectives that deal with individual and societal risks posed by nuclear power plant operation. The objective of the policy statement was to establish goals that broadly define an acceptable level of radiological risk that might be imposed on the public as a result of nuclear power plant operation.

The Commission recognized that the safety goals could provide a useful tool by which the adequacy of regulations or regulatory decisions regarding changes to the regulations could be judged. Safety goals could be of benefit also in the much more difficult task of assessing whether existing plants that have been designed, constructed, and operated to comply with past and current regulations conform adequately with the intent of the safety goal policy.

The models and results of NUREG–1150 can be used in a number of ways in the NRC staff's analysis and implementation of safety goal policy. For example, the five plants studied for this report have been compared with the two quantitative health objectives, as shown in Figure 13.2 for internal initiators. Figure 13.3 compares Surry and Peach Bottom with the quantitative health objectives for fire initiators. As may be seen, the present risk estimates for these five plants (considering internally initiated accidents) and for the Surry and Peach Bottom plants (considering fire initiators) fall beneath the quantitative health objective risk goals. In addition, however, it may be seen that the risk estimates among the five plants vary considerably. An analysis of the plant design and operational differences that cause this variability can provide valuable information to the staff in its consideration of the balance of the present set of regulations and the areas of regulation that could most benefit from improvement.

The staff has reviewed the NUREG–1150 results at a broad level to determine the causes of the variability among plant risks shown in Figure 13.2. A number of design, operational, and siting factors are important to this measure of plant risk and determine the relative location of a specific plant's risk range in comparison with other plants and with the safety goal. At a general level, core damage frequency, containment and source term performance, and surrounding population demographics all can affect the risk range. Thus, using the Surry plant as an example, the combination of a relatively low core damage frequency, relatively good containment performance, and a low population density act to ensure with a high probability that the risk is below the safety goal.

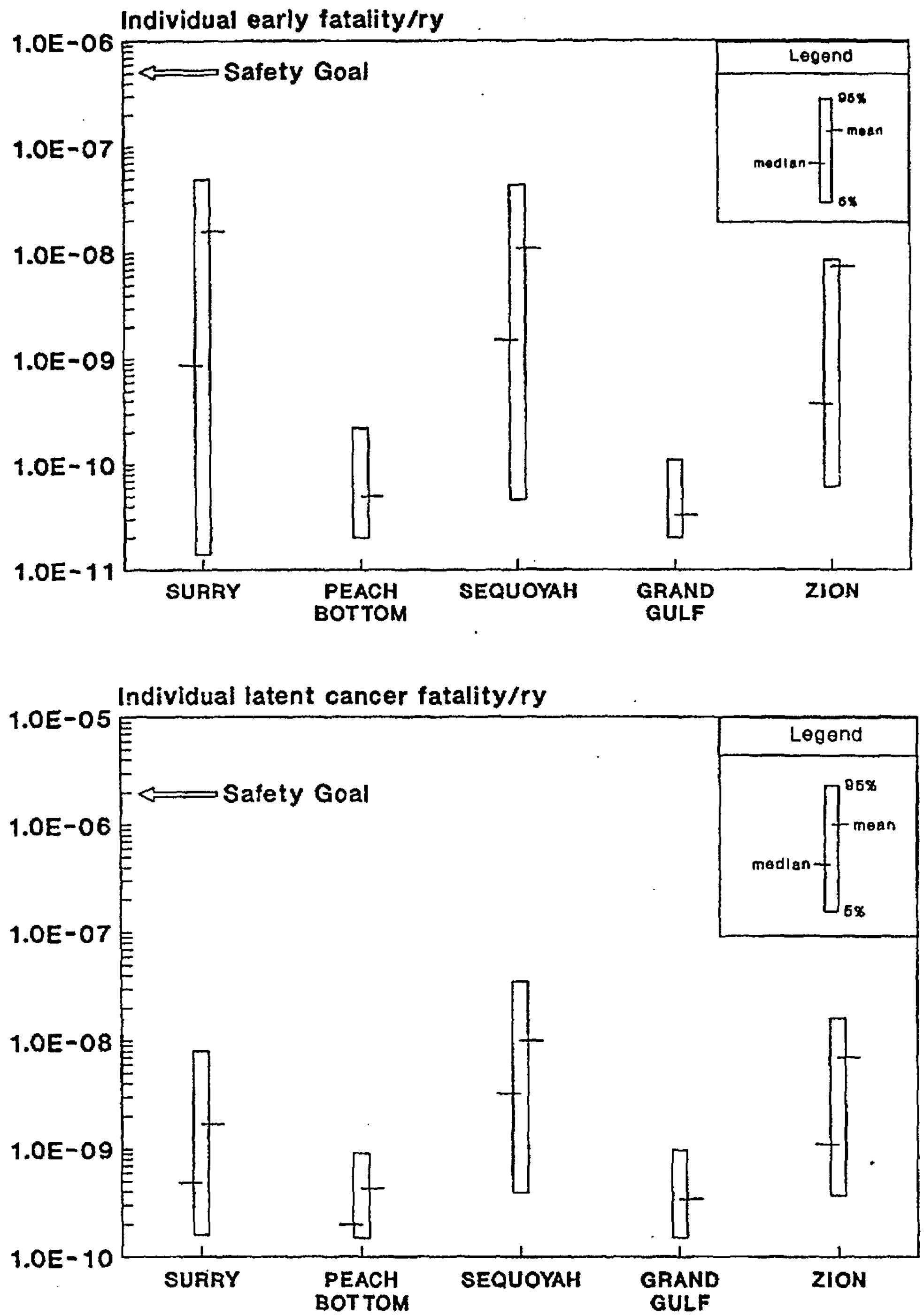
The NUREG–1150 results can also be used to support the analysis of alternative safety goal implementation approaches. One subject of discussion in the staff's work is the need for a supplemental definition of containment performance in severe accidents using the probability of a large release as a measure. An acceptable frequency for such a release was defined as 1E–6 per reactor year. A potential definition of a large release is one that can cause one or more early fatalities.* The present NUREG–1150 risk analyses have been evaluated to provide the frequency of such a release, as shown in Figure 13.4. The mean large release probabilities are below 1E–6 per reactor year. Further staff work in assessing alternative definitions is planned as part of the safety goal implementation program, and it is expected that NUREG–1150 methods and results will be used.

13.2.6 Effect of Emergency Preparedness on Consequence Estimates

NUREG–1150 provides information for developing protective action strategies that could be followed near a nuclear power plant in case of a severe accident. In developing strategies, consideration must be given to several types of protective actions, such as sheltering, evacuation, and relocation and various combinations. These strategies are influenced by the types of severe accidents that might occur at a nuclear power plant, their frequency of occurrence, and the radioactive release expected to result from each accident type as well as by the topography, weather, population density, and other site-specific characteristics.

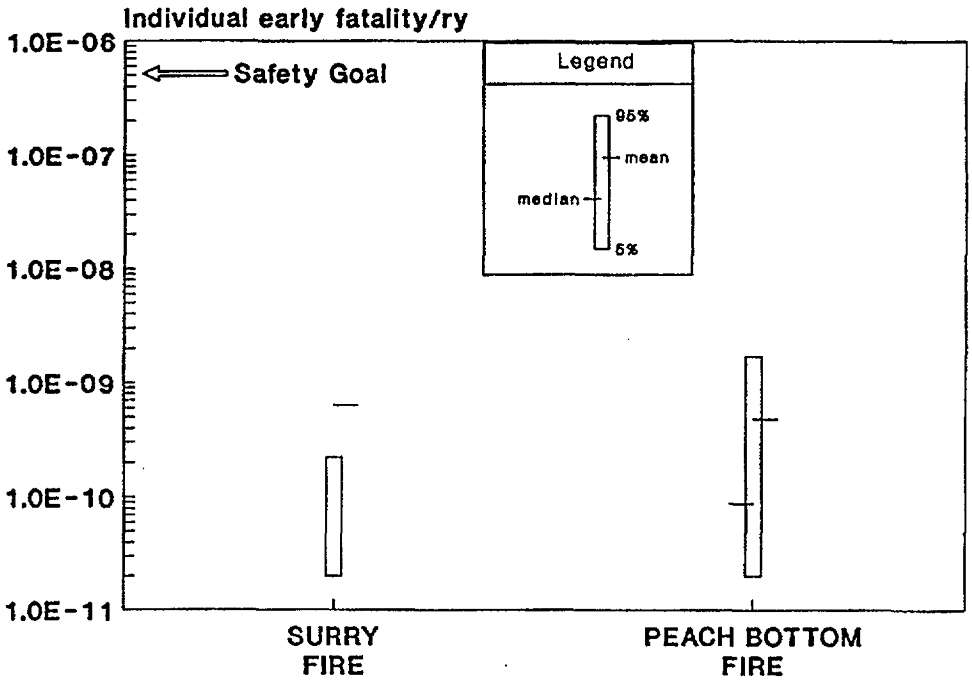
NUREG–1150 provides assessments of a broad spectrum of potential core damage accidents that could occur at a nuclear power plant. These assessments permit the evaluation of hypothetical

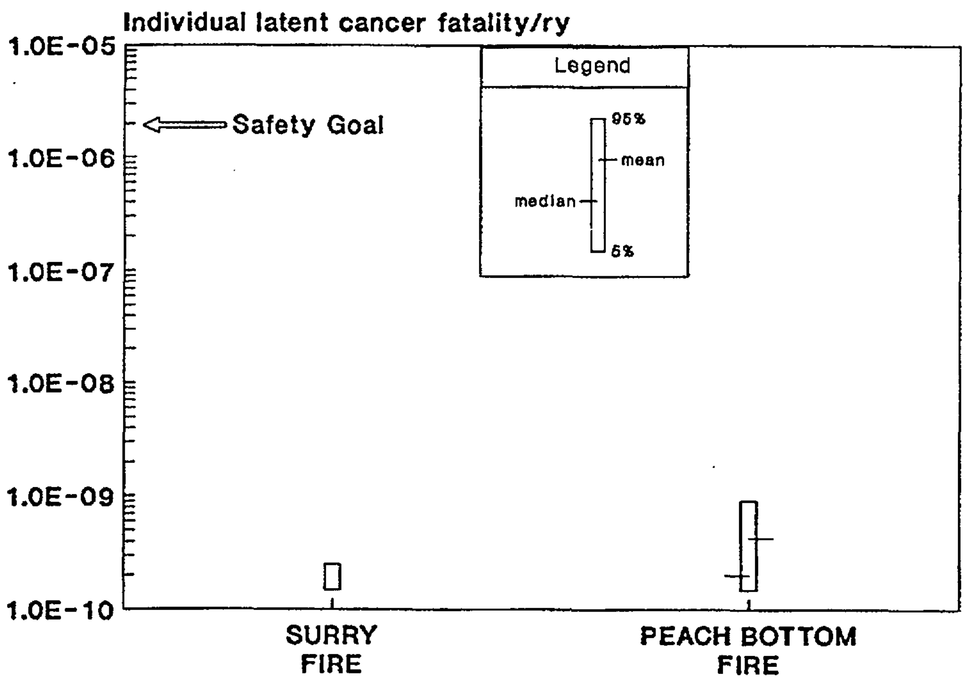
*The Commission has now indicated that this is not an appropriate definition and has asked the staff to review and propose an alternative definition.



Note: As discussed in Reference 13.23, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

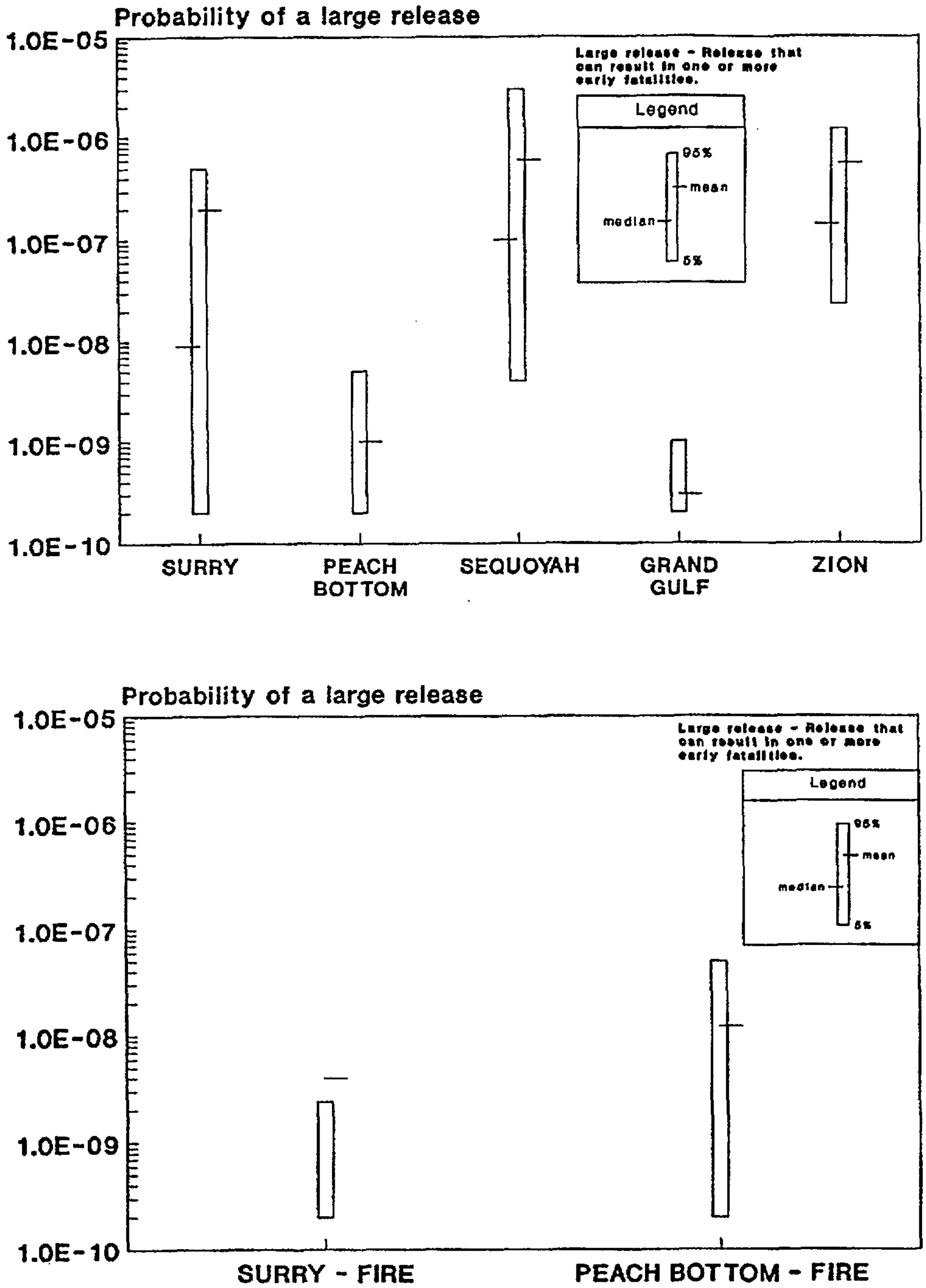
Figure 13.2 Comparison of individual early and latent cancer fatality risks at all plants (internal initiators).





Note: As discussed in Reference 13.23, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 13.3 Comparison of individual early and latent cancer fatality risks at Surry and Peach Bottom (fire initiators).



Note: As discussed in Reference 13.23, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 13.4 Frequency of one or more early fatalities.

dose savings for a spectrum of accidents and provide a means for evaluating potential reduction in early severe health effects (injuries and fatalities) in the event of an accident by implementing emergency response strategies.

The most important considerations in establishing emergency preparedness strategies are the warning times before release to initiate the emergency response and magnitude of the release of the radioactive material to the environment. The warning time and magnitude of radioactive release are in turn strongly influenced by the time and size of containment failure or bypass. If the containment fails early, the radioactive release is generally larger and more difficult to predict than if the containment fails late.

To evaluate the effectiveness of various protective actions, the conditional probabilities of acute red bone marrow doses exceeding 200 rems and 50 rems were calculated for several possible actions, using Zion plant source terms as examples. Doses were calculated on the plume centerline for various distances from the plant. The actions evaluated are:

- Normal activity—assumed that no protective actions were taken during the release but assumed that people were relocated within 6 hours of plume arrival.
- Home sheltering—sheltering in a single family home (see Table 11.5 for a definition of sheltering). The penetration fractions for groundshine and cloudshine were representative of masonry houses without basements as well as wood frame houses with basements. Indoor protection for inhalation of radionuclides was assumed. People were relocated from the shelter mode within 6 hours of plume arrival.
- Large building shelter—sheltering in a large building, for example, an office building, hospital, apartment building, or school. Indoor protection for inhalation of radionuclides was assumed. People were relocated from the shelter mode within 6 hours of plume arrival.
- Evacuation—doses were calculated for people starting to travel at the time of release, 1 hour before start of release, and 1 hour after start of release. An evacuation speed of 2.5 mph was assumed.

Figure 13.5 shows the conditional probabilities of exceeding a 50-rem and a 200-rem red bone marrow dose for the various possible response modes assuming an early containment failure at Zion with source term magnitudes varying from low to high. Figure 13.6 shows similar results for a late containment failure at Zion.

Use of the above assumptions indicates that if a large release occurs (Fig. 13.5), there is a large probability of doses exceeding 200 rems within 1 to 2 miles from the reactor. Sheltering does not significantly lower this probability. Thus, if a large release can occur, it is prudent to consider prompt evacuation prior to the start of the release.

At 3 miles and beyond, it is possible to avoid doses exceeding 200 rems by sheltering in large buildings even if a large release were to occur. Thus, people in large buildings such as hospitals would not necessarily have to be immediately evacuated, but could shelter instead. Of course, further reductions in dose are possible by evacuation.

At 10 miles, no protective actions except relocation would be necessary to avoid 200-rem doses. Sheltering in large buildings or evacuation prior to release would probably keep doses below 50 rems.

13.3 Major Factors Contributing to Risk

NUREG–1150 results can be used to identify dominant plant risk contributors and associated uncertainties. A discussion of these dominant risk contributors is found in Chapters 3 through 8 and Chapter 12. This section focuses on the use in guiding research, generic issue resolution, and inspection programs.

Because of its integrated nature, discussion of uncertainties, and reliance on more realistic assessments, PRA-based information found in NUREG–1150 and its supporting documents can be used to guide and focus a wide spectrum of activities designed to improve the state of knowledge regarding the safety of individual nuclear power plants, as well as that of the nuclear industry as a whole. The resources of both the NRC and the industry are limited, and the application of PRA techniques and subsequent insights provides an important tool to aid the decisionmaker in effectively allocating these resources.

The nature of the many decisions necessary to allocate regulatory resources does not require great precision in PRA results. For example, in assigning priorities to research or efforts to resolve generic safety issues, it is sufficient to use broad

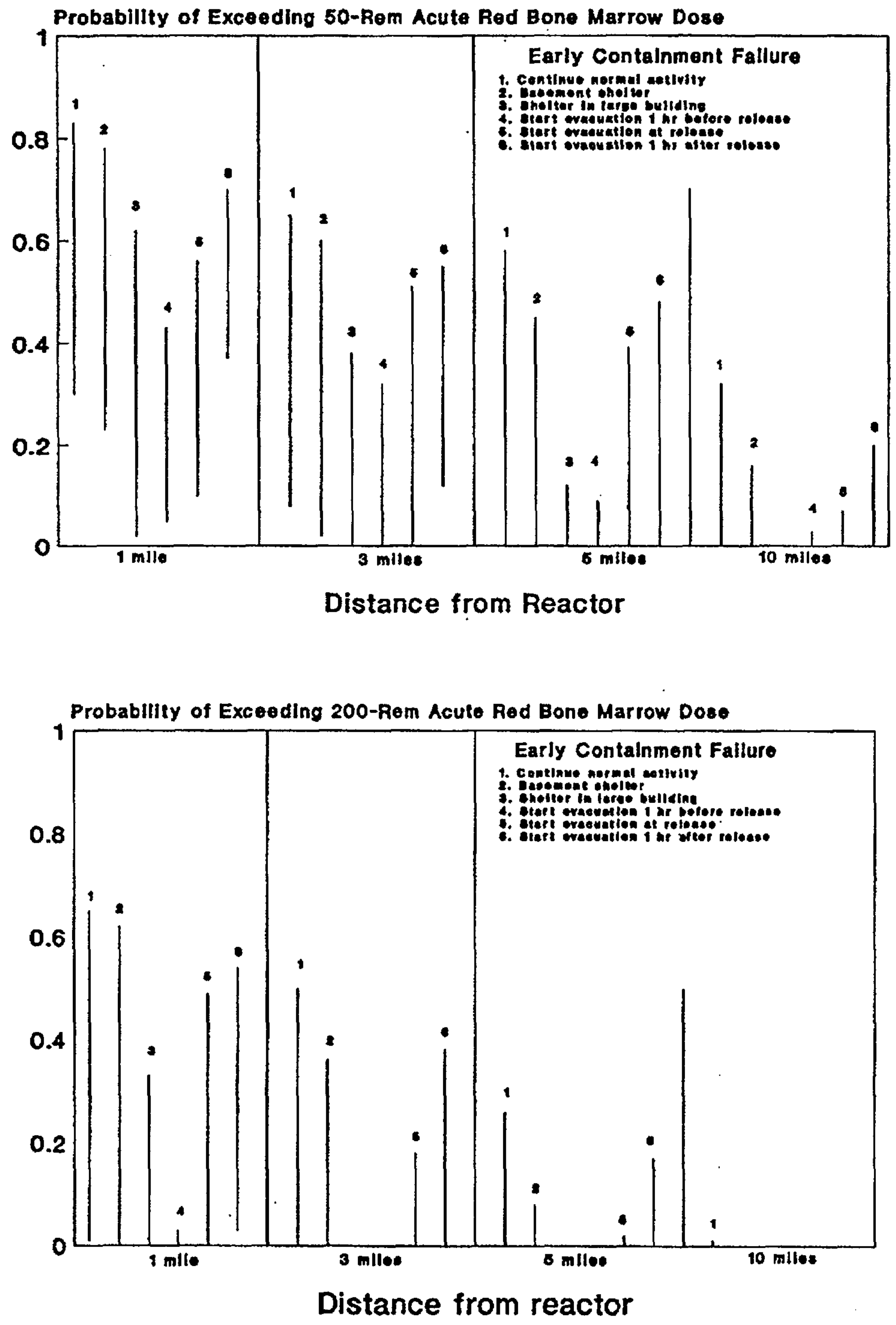
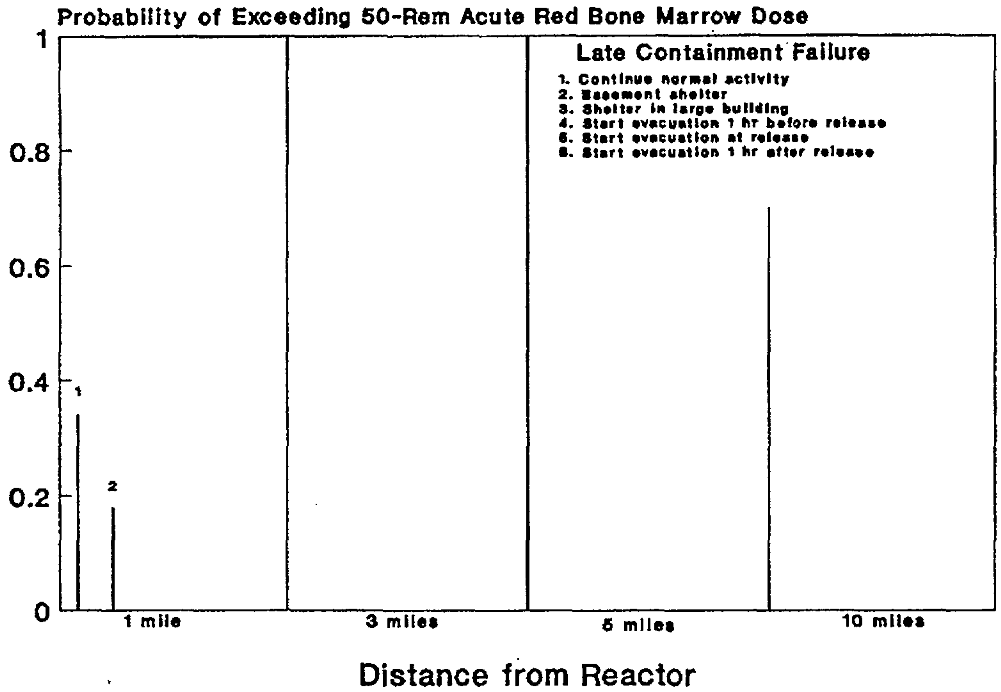


Figure 13.5 Relative effectiveness of emergency response actions assuming early containment failure with high and low source terms.



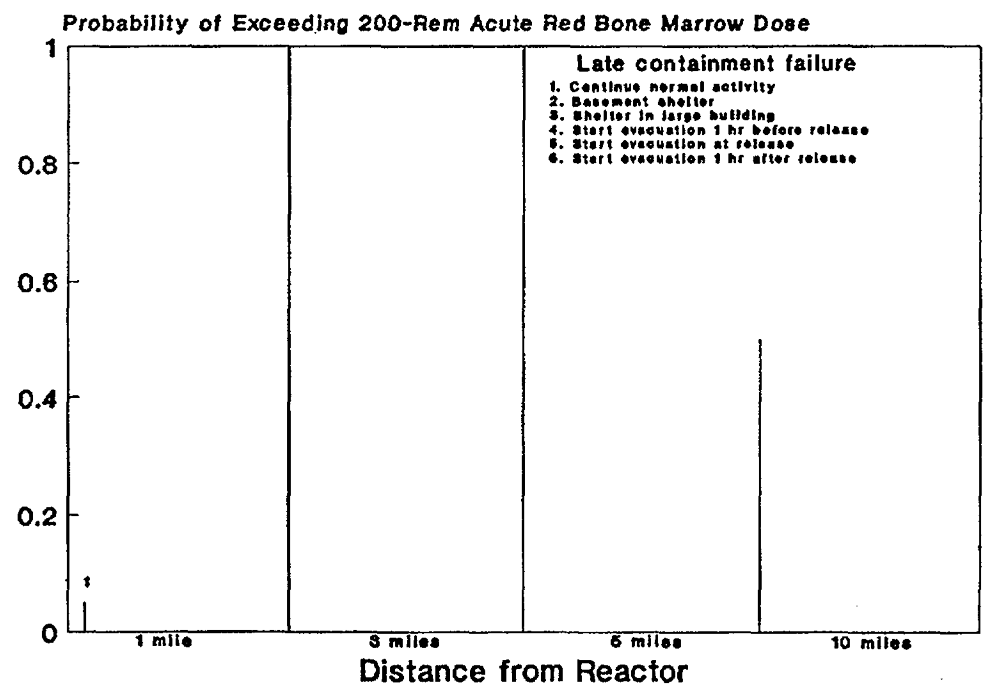


Figure 13.6 Relative effectiveness of emergency response actions assuming late containment failure with high and low source terms.

categories of risk impact (e.g., high, medium, and low) (Ref. 13.24). In a similar manner, information from PRAs can be used to guide the allocation of resources in inspection and enforcement programs (see Section 13.3.3).

13.3.1 Reactor Research

As noted earlier, the nature of the decisions necessary to allocate resources does not require great precision in PRA results. In prioritizing research efforts, it is sufficient to use broad categories of risk impact (e.g., high, medium, and low). A given issue can be evaluated in terms of the number of plants affected, the risk impacts on each plant, the effect of modifications in reducing the risk, and the effect of additional knowledge on improving the prediction of plant risk or severe core damage frequency or on reducing or defining more clearly the associated uncertainties. These generic measures of significance, combined appropriately with other information (e.g., cost of resolving the issue) can be used to evaluate the issue under consideration.

13.3.2 Prioritization of Generic Issues

The NRC has been setting priorities for generic safety issues for several years using PRA as one informational input (Ref. 13.25). In prioritizing efforts to resolve generic safety issues, it is sufficient to use broad categories of risk impact (e.g., high, medium, and low) in which only order-of-magnitude variations are considered important. The reasoning is that a potential safety issue would not be dismissed unless it were clearly of low risk. Thus, one or more completed PRA studies can often be selected as surrogates for the purpose of assigning such priorities, even though they clearly do not fully represent the characteristics of some plants, provided the nature of the difference is reasonably understood and can be qualitatively evaluated.

As with any priority-assignment method, the final results must be tempered with an engineering evaluation of the reasonableness of the assignment, and the PRA-based analysis can serve as only one ingredient of the overall decision.

One of the most important benefits of using PRA as an aid to assigning priorities is the documentation of a comprehensive and disciplined analysis of the issue, which enhances debate on the merits of specific aspects of the issue and reduces reliance on more subjective judgments. Clearly, some issues would be very difficult to quantify with reasonable accuracy, and the assignment of priorities to these issues would have to be based largely on subjective judgment.

PRA is being usefully applied to setting priorities for generic safety issues and to evaluating new issues as they are identified. In this effort, each issue is assessed as to its nature, its probable core damage frequency and public risk, and the cost of one or more conceptual fixes that could resolve the issue. A matrix is developed whereby each issue is characterized as of high, medium, or low probability, or whether the issue should be summarily dropped from further regulatory consideration. This matrix considers both the absolute magnitude of the core damage frequency or risk and the value/impact ratio of conceptual fixes. Risk-reduction estimates are normally made using surrogate PWRs and BWRs, based on existing PRAs.

A principal benefit of PRA-based prioritization, compared to other methods for allocating resources to safety issues, is that important assumptions made in quantifying the risk are displayed and uncertainties in the analyses are estimated. A principal limitation is that some of the issues, such as those dealing with human factors, are only subjectively quantified. Thus, the uncertainties can be large. However, on balance, PRA-based prioritization has been found to be quite useful. Although uncertainties may be large, the process forces attention on these uncertainties to a much higher degree than if the quantification were not attempted. Also, the uncertainties are normally part of the issues themselves and not just an artifact of the PRA analysis.

Since, as discussed above, the prioritization is done on an approximate (order-of-magnitude) basis, the new information developed in NUREG-1150 is not expected to substantially change previously developed priority rankings. However, a sample of key issues will be reexamined to determine whether, based on the updated information in NUREG-1150, changes in dominant accident sequences or performance of mitigative systems could substantially affect the previous rankings.

13.3.3 Use of PRA in Inspections

The importance to NRC of risk-based inspection data is exemplified by the following statement in NRC's 5-Year Plan: "Probabilistic risk assessment techniques will be applied to all phases of the inspection program in order to insure that inspection activities are prioritized and conducted in an integrated fashion." Within NRC, the Risk Applications Branch of the Office of Nuclear Reactor Regulation has the responsibility of directly providing risk-based information to the regional offices and resident inspectors. This ongoing effort has resulted in the development of plant-specific, and in some cases generic, PRA perspectives that help to provide an optimization of inspection resources and a prioritization of inspection resources on the high-risk aspects of a plant. Using draft NUREG–1150 data, team inspection procedures based on plant-specific PRA information have been developed and implemented on such plants as Grand Gulf. Formalization of these inspection activities can be found in a recently issued inspection module entitled "Risk Focused Operation Readiness Inspection Procedures." This module focuses on how to use PRA perspectives and conduct a risk-based team inspection based on risk insights. The spectrum of reactor plant design types addressed in NUREG–1150 provide a broad risk data base that in many instances can be used to assist in inspection-type decisions even for plants without a PRA.

REFERENCES FOR CHAPTER 13

13.1 U.S. Nuclear Regulatory Commission (USNRC), "Integration Plan for Closure of Severe Accident Issues," SECY-88-147, May 25, 1988.

13.2 USNRC, "Reactor Safety Study—An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

13.3 D. M. Ericson, Jr.,(Ed.) et al., "Analysis of Core Damage Frequency: Internal Events Methodology," Sandia National Laboratories, NUREG/CR-4550, Vol. 1, Revision 1, SAND86-2084, January 1990.

13.4 T. A. Wheeler et al., "Analysis of Core Damage Frequency from Internal Events: Expert Judgment Elicitation," Sandia National Laboratories, NUREG/CR-4550, Vol. 2, SAND86-2084, April 1989.

13.5 R. C. Bertucio and J. A. Julius, "Analysis of Core Damage Frequency: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 3, Revision 1, SAND86-2084, April 1990.

13.6 A. M. Kolaczkowski et al., "Analysis of Core Damage Frequency: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4550, Vol. 4, Revision 1, SAND86-2084, August 1989.

13.7 R. C. Bertucio and S. R. Brown, "Analysis of Core Damage Frequency: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 5, Revision 1, SAND86-2084, April 1990.

13.8 M. T. Drouin et al., "Analysis of Core Damage Frequency: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 6, Revision 1, SAND86-2084, September 1989.

13.9 M. B. Sattison and K. W. Hall, "Analysis of Core Damage Frequency: Zion Unit 1," Idaho National Engineering Laboratory, NUREG/CR-4550, Vol. 7, Revision 1, EGG-2554, May 1990.

13.10 E. D. Gorham-Bergeron et al., "Evaluation of Severe Accident Risks: Methodology for the Accident Progression, Source Term, Consequence, Risk Integration, and Uncertainty Analyses," Sandia National Laboratories, NUREG/CR-4551, Vol. 1, Draft Revision 1, SAND86-1309, to be published.*

13.11 F. T. Harper et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," Sandia National Laboratories, NUREG/CR-4551, Vol. 2, Revision 1, SAND86-1309, December 1990.

13.12 R. J. Breeding et al., "Evaluation of Severe Accident Risks: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 3, Revision 1, SAND86-1309, October 1990.

13.13 A. C. Payne, Jr., et al., "Evaluation of Severe Accident Risks: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4551, Vol. 4, Draft Revision 1, SAND86-1309, to be published.*

13.14 J. J. Gregory et al., "Evaluation of Severe Accident Risks: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 5, Revision 1, SAND86-1309, December 1990.

13.15 T. D. Brown et al., "Evaluation of Severe Accident Risks: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 6, Draft Revision 1, SAND86-1309, to be published.*

13.16 C. K. Park et al., "Evaluation of Severe Accident Risks: Zion Unit 1," Brookhaven National Laboratory, NUREG/CR-4551, Vol. 7, Draft Revision 1, BNL-NUREG-52029, to be published.*

13.17 USNRC, "Individual Plant Examination for Severe Accident Vulnerabilities—10 CFR §50.54(f)," Generic Letter 88-20, November 23, 1988.

13.18 USNRC, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, Draft Report for Comment, July 1990.

*Available in the NRC Public Document Room, 2120 L Street NW., Washington, DC.

13.19 USNRC, "Probabilistic Risk Assessment Reference Document," NUREG-1050, September 1984.

13.20 A. L. Camp et al., "The Risk Management Implications of NUREG-1150 Methods and Results," Sandia National Laboratories, NUREG/CR-5263, SAND88-3100, September 1989.

13.21 M. T. Drouin et al., "Analysis of Core Damage Frequency from Internal Events: Grand Gulf, Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 6, SAND86-2084, April 1987.

13.22 USNRC, "Safety Goals for the Operation of Nuclear Power Plants; Policy Statement," *Federal Register*, Vol. 51, p. 30028, August 21, 1986.

13.23 H. J. C. Kouts et al., "Special Committee Review of the Nuclear Regulatory Commission's Severe Accident Risks Report (NUREG-1150)," NUREG-1420, August 1990.

13.24 USNRC, "Categorization of Reactor Safety Issues from a Risk Perspective," NUREG-1115, March 1985.

13.25 R. Emrit et al., "A Prioritization of Generic Safety Issues," USNRC Report NUREG-0933, July 1984.

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11. ABSTRACT (200 words or less)

This report summarizes an assessment of the risks from severe accidents in five commercial nuclear power plants in the United States. These risks are measured in a number of ways, including: the estimated frequencies of core damage accidents from internally initiated accidents, and externally initiated accidents for two of the plants; the performance of containment structures under severe accident loadings; the potential magnitude of radionuclide releases and offsite consequences of such accidents; and the overall risk (the product of accident frequencies and consequences). Supporting this summary report are a large number of reports written under contract to NRC which provide the detailed discussion of the methods used and results obtained in these risk studies.

Volume 1 of this report has three parts. Part I provides the background and objectives of the assessment and summarizes the methods used to perform the risk studies. Part II provides a summary of results obtained for each of the five plants studied. Part III provides perspectives on the results and discusses the role of this work in the larger context of the NRC staff's work.

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