PART II

Summary of Plant Results

3. SURRY PLANT RESULTS

3.1 Summary Design Information

The Surry Power Station is a two-unit site. Each unit, designed by the Westinghouse Corporation, is a three-loop pressurized water reactor (PWR) rated at 2441 MWt (788 MWe) and is housed in a subatmospheric containment designed by Stone and Webster Engineering Corporation. The balance of plant systems were engineered and built by Stone and Webster Engineering Corporation. Located on the James River near Williamsburg, Virginia, Surry 1 started commercial operation in 1972. Some important system design features of the Surry plant are described in Table 3.1. A general plant schematic is provided in Figure 3.1.

This chapter provides a summary of the results obtained in the detailed risk analyses underlying this report (Refs. 3.1 and 3.2). A discussion of perspectives with respect to these results is provided in Chapters 8 through 12.

3.2 Core Damage Frequency Estimates

3.2.1 Summary of Core Damage Frequency Estimates

The core damage frequency and risk analyses performed for this study considered accidents initiated by both internal and external events (Ref. 3.1). The core damage frequency results obtained from internal events are provided in graphical form, displayed as a histogram, in Figure 3.2 (Section 2.2.2 discusses histogram development). The core damage frequency results obtained from both internal and external events are provided in tabular form in Table 3.2.

The Surry plant was previously analyzed in the Reactor Safety Study (RSS) (Ref. 3.3). The RSS calculated a point estimate core damage frequency from internal events of 4.6E–5 per year. The present study calculated a total median core damage frequency from internal events of 2.3E–5 per year. For a detailed discussion of, and insights into, the comparison between this study and the RSS, see Chapter 8.

3.2.1.1 Internally Initiated Accident Sequences

A detailed description of accident sequences important at the Surry plant is provided in Reference 3.1. For this summary report, the accident sequences described in that report have been grouped into five summary plant damage states. These are:

- Station blackout,
- Large and small loss-of-coolant accidents (LOCAs),
- Anticipated transients without scram (ATWS),
- All other transients except station blackout and ATWS, and
- Interfacing-system LOCA and steam generator tube rupture.

The relative contributions of these groups to the mean internal-event core damage frequency at Surry are shown in Figure 3.3. From Figure 3.3, it is seen that station blackout sequences are the largest contributors to mean core damage frequency. It should be noted that the plant configuration was modeled as of March 1988 and thus does not reflect implementation of the station blackout rule.

Within the general class of station blackout accidents, the more probable combinations of failures leading to core damage are:

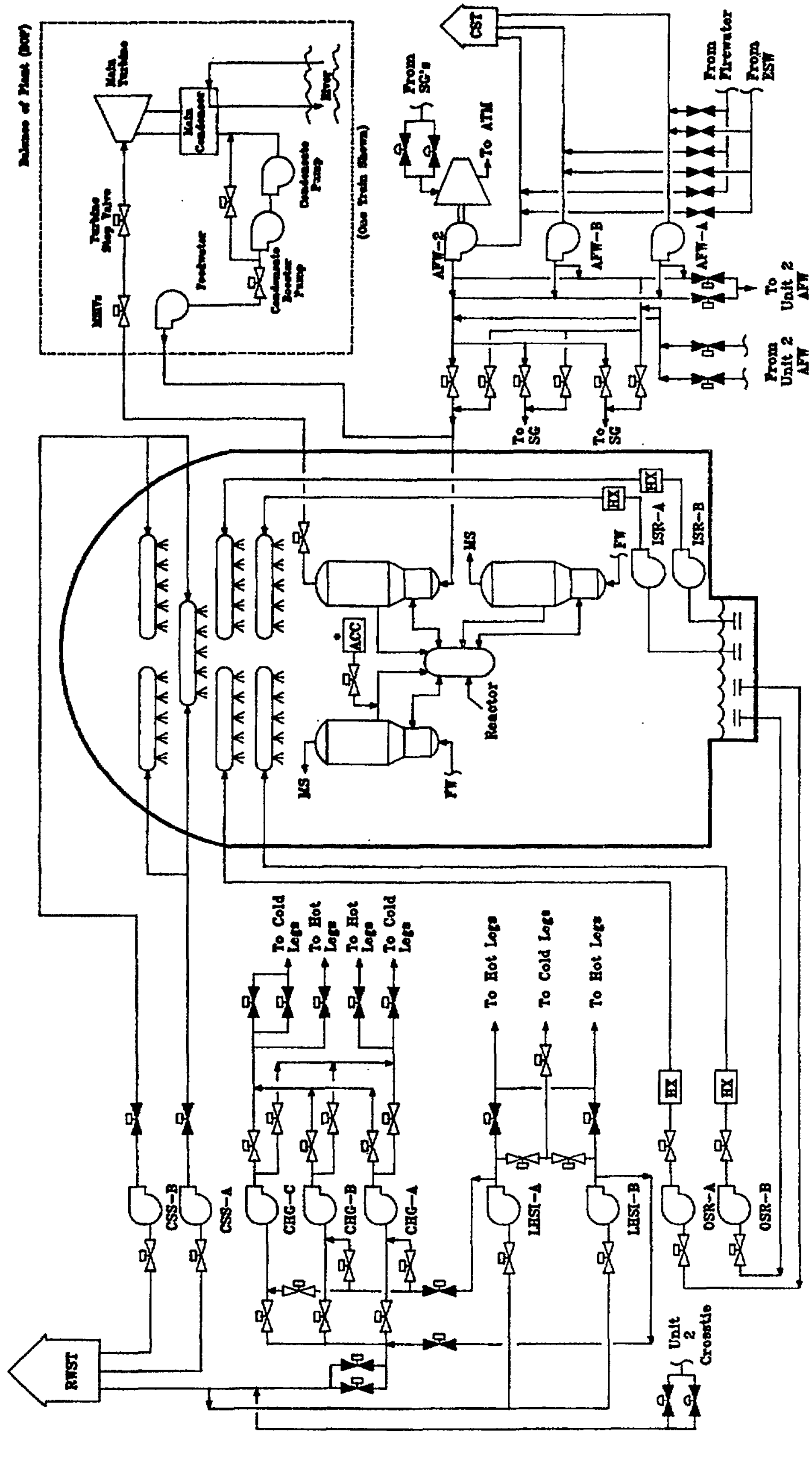
- Loss of onsite and offsite ac power and failure of the auxiliary feedwater (AFW) system. All core heat removal is unavailable after failure of AFW. Station blackout results in the unavailability of the high-pressure injection system, the containment spray system, and the inside and outside containment spray recirculation systems. For station blackout at Unit 1 alone, it was assessed that one high-pressure injection (HPI) pump at Unit 2 would not be sufficient to provide feed and bleed cooling through the crossconnect while at the same time provide charging flow to Unit 2. Core damage was estimated to begin in approximately 1 hour if AFW and HPI flow had not been restored by that time.

- Loss of onsite and offsite ac power results in the unavailability of the high-pressure injection system, the containment spray system, the inside and outside containment spray recirculation systems, and the motor-driven auxiliary feedwater pumps. While the loss of all ac power does not affect instrumentation at the start of the station blackout, a long

Table 3.1 Summary of design features: Surry Unit 1.

1. Coolant Injection Systems	a. High-pressure safety injection and recirculation system with 2 trains and 3 pumps. b. Low-pressure injection and recirculation system with 2 trains and 2 pumps. c. Charging system provides normal makeup flow with safety injection crosstie to Unit 2.
2. Steam Generator Heat Removal Systems	a. Power conversion system. b. Auxiliary feedwater system (AFWS) with 3 trains and 3 pumps (2 MDPs, 1 TDP)* and crosstie to Unit 2 AFWS.
3. Reactivity Control Systems	a. Control rods. b. Chemical and volume control systems.
4. Key Support Systems	a. dc power provided by 2-hour design basis station batteries. b. Emergency ac power provided by 1 dedicated and 1 swing diesel generator (both self-cooled). c. Component cooling water provides cooling to RCP thermal barriers. d. Service water is gravity-fed system that provides heat removal from containment following an accident.
5. Containment Structure	a. Subatmospheric (10 psia). b. 1.8 million cubic feet. c. 45 psig design pressure. d. Reinforced concrete.
6. Containment Systems	a. Spray injection initiated at 25 psia with 2 trains and 2 pumps. b. Inside spray recirculation initiated (with 2-minute time delay) at 25 psia with 2 trains and 2 pumps (both pumps inside containment). c. Outside spray recirculation initiated (with 5-minute time delay) at 25 psia with 2 trains and 2 pumps (both pumps outside containment). d. Inside and outside spray recirculation systems are the only sources of containment heat removal after a LOCA.

*MDP — Motor-Driven Pump.
TDP — Turbine-Driven Pump.



*Typical of each Cold Leg Loop

Figure 3.1 Surry plant schematic.

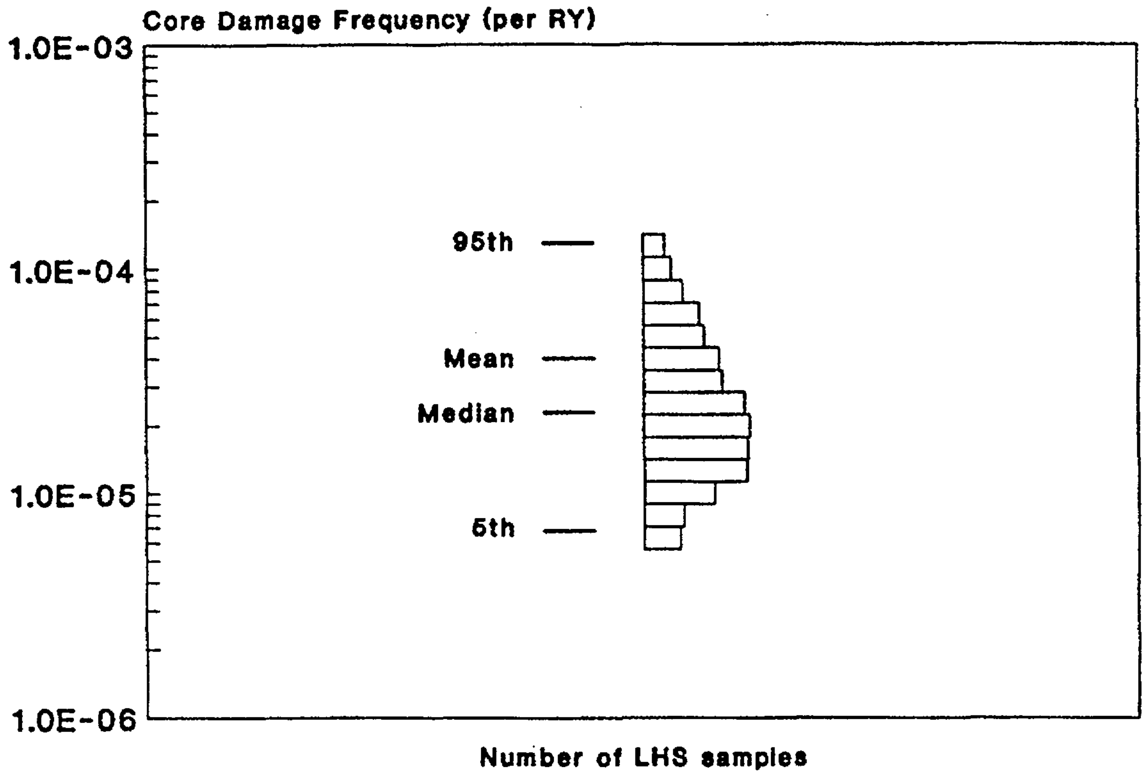


Figure 3.2 Internal core damage frequency results at Surry.*

Table 3.2 Summary of core damage frequency results: Surry.*

	5%	Median	Mean	95%
Internal Events	6.8E-6	2.3E-5	4.0E-5	1.3E-4
Station Blackout				
Short Term	1.1E-7	1.7E-6	5.4E-6	2.3E-5
Long Term	6.1E-7	8.2E-6	2.2E-5	9.5E-5
ATWS	3.2E-8	4.2E-7	1.6E-6	5.9E-6
Transient	7.2E-8	6.9E-7	2.0E-6	6.0E-6
LOCA	1.2E-6	3.8E-6	6.0E-6	1.6E-5
Interfacing LOCA	3.8E-11	4.9E-8	1.6E-6	5.3E-6
SGTR	1.2E-7	7.4E-7	1.8E-6	6.0E-6
External Events**				
Seismic (LLNL)	3.9E-7	1.5E-5	1.2E-4	4.4E-4
Seismic (EPRI)	3.0E-7	6.1E-6	2.5E-5	1.0E-4
Fire	5.4E-7	8.3E-6	1.1E-5	3.8E-5

*As discussed in Reference 3.4, 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).

**See "Externally Initiated Accident Sequences" in Section 3.2.1.2 for discussion.

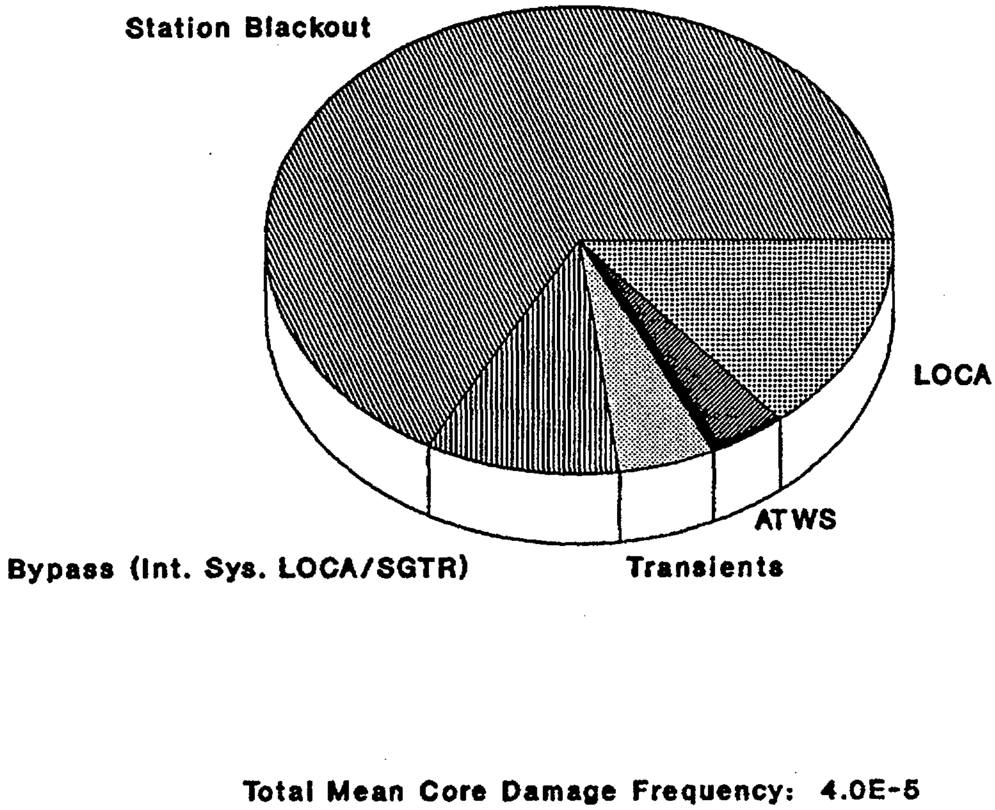


Figure 3.3 Contributors to mean core damage frequency from internal events at Surry.

duration station blackout leads to battery depletion and subsequent loss of vital instrumentation. Battery depletion was concluded to occur after approximately 4 hours. The ability to subsequently provide decay heat removal with the turbine-driven AFW pump is lost because of the loss of all instrumentation and control power. Using information from Reference 3.5, approximately 3 hours beyond the time of battery depletion was allowed for restoration of ac power before core uncovery would occur.

- Loss of onsite and offsite ac power, followed by a reactor coolant pump seal LOCA due to loss of all seal cooling. Station blackout also results in the unavailability of the HPI system, as well as the auxiliary feedwater motor-driven pumps, the containment spray system, and the inside and outside spray recirculation systems. Continued coolant loss through the failed seals, with unavailability of the HPI system, leads to core uncovery.

Within the general class of LOCAs, the more probable combinations of failures are:

- LOCA with an equivalent diameter of greater than 6 inches in the reactor coolant system (RCS) piping with failure of the low-pressure injection or recirculation system. Recovery of equipment is unlikely for the system failures assessed to be most likely and, because the break size is sufficiently large, the time to core uncovery is approximately 5 to 10 minutes, leaving virtually no time for recovery actions. All containment heat removal systems are available. The dominant contributors to failure of the low-pressure recirculation function are the common-cause failure of the refueling water storage tank (RWST) isolation valves to close, common-cause failure of the pump suction valves to open, common-cause failure of the discharge isolation valves to the hot legs to open, or miscalibration of the RWST level sensors.

- Intermediate-size LOCAs with an equivalent diameter of between 2 and 6 inches in the

RCS piping with failure of the low-pressure injection or recirculation core cooling system. All containment heat removal systems are available, but the continued heatup and boiloff of primary coolant leads to core uncovery in 20 to 50 minutes. The dominant contributors to low-pressure injection failure are common-cause failure of the low-pressure injection (LPI) pumps to start or plugging of the normally open LPI injection valves.

- Small-size LOCAs with an equivalent diameter of between 1/2 and 2 inches in the RCS piping with failure of the HPI system. All containment heat removal systems are available, but the continued heatup and boiloff of primary coolant leads to core uncovery in 1 to 8 hours. The dominant contributors to HPI system failures are hardware failures of the check valves in the common suction and discharge line of all three charging pumps or common-cause failure of the motor-operated valves in the HPI discharge line.

Within the general class of containment bypass accidents, the more probable combinations of failures are:

- An interfacing-system LOCA resulting from a failure of any one of the three pairs of check valves in series that are used to isolate the high-pressure RCS from the LPI system. The failure modes of interest for Event V are rupture of valve internals on both valves or failure of one valve to close upon repressurization (e.g., during a return to power from cold shutdown) combined with rupture of the other valve. The resultant flow into the low-pressure system is assumed to result in failure (rupture) of the low-pressure piping or components outside the containment boundary. Although core inventory makeup by the high-pressure systems is initially available, inability to switch to recirculation would eventually lead to core damage approximately 1 hour after the initial failure. Because of the location of the postulated system failure (outside containment), all containment mitigating systems are bypassed.

- A steam generator tube rupture (SGTR) accident initiated by the double-ended guillotine rupture of one steam generator (SG) tube. (Multiple tube ruptures may be possible but were not considered in this analysis.) If the operators fail to depressurize the reactor coolant system in a timely manner (in about 45 minutes), there is a high probability that water will be forced through the safety relief valves (SRVs) on the steam line from the affected SG. The probability that the SRVs will fail to reclose under these conditions is also estimated to be very high (near 1.0). Failure to close (gag the SRVs) by a local, manual action results in a non-isolable path from the RCS to the environment. After the entire contents of the refueling water storage tank are pumped through the broken SG tube, the core uncovers. The onset of core degradation is thus not expected until about 10 hours after the start of the accident.

3.2.1.2 Externally Initiated Accident Sequences

A detailed description of accident sequences initiated by external events important at the Surry plant is provided in Part 3 of Reference 3.1. The accident sequences described in that reference have been divided into two main types for this study. These are:

- Seismic, and
- Fire.

A scoping study has also been performed to assess the potential effects of other externally initiated accidents (Ref. 3.1, Part 3). This analysis indicated that the following external-event sources could be excluded based on the low frequency of the initiating event:

- Air crashes,
- Hurricanes,
- Tornados,
- Internal flooding, and
- External flooding.

1. Seismic Accident Frequency Analysis

The relative contribution of classes of seismically and fire-initiated accidents to the total mean frequency of externally initiated core damage accidents is provided in Figure 3.4. As may be seen, seismically initiated loss of offsite power plant transients and transients that (through cooling system failures) lead to reactor coolant pump seal LOCAs are the most likely causes of externally caused core damage accidents. For these two accident initiators, the more probable combinations of system failures are:

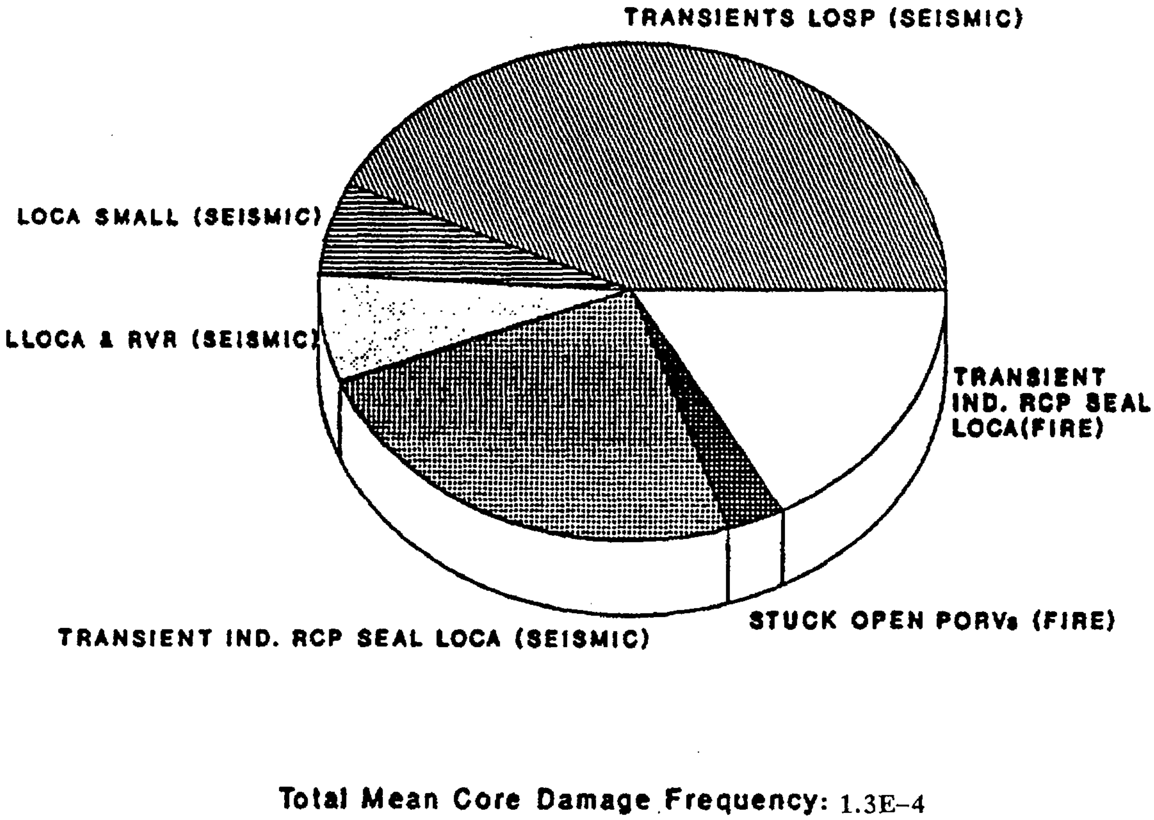


Figure 3.4 Contributors to mean core damage frequency from external events (LLNL hazard curve) at Surry.

- Transient-initiated accident sequences resulting from loss of offsite power in conjunction with failures of the auxiliary feedwater system and failure of the feed and bleed mode of core cooling. These result from either seismically induced diesel generator failures (causing station blackout and eventual battery depletion) or from seismically induced failure of the condensate storage tank in conjunction with power-operated relief valve (PORV) failures.

- Loss of offsite power (LOSP) due to seismically induced failure of ceramic insulators in the switchyard, with simultaneous (seismic) failure of both high-pressure injection (HPI) and component cooling water (CCW) systems (the redundant sources of seal cooling). Failures of HPI result from seismic failures of the refueling water storage tank or emergency diesel generator load panels, while seismic failures of the diesels or the CCW heat exchanger supports result in loss of the CCW system.

As discussed in Chapter 2, the seismic analysis in this report made use of two sets of hazard curves from Lawrence Livermore National Laboratory (LLNL) (Ref. 3.6) and the Electric Power Research Institute (EPRI) (Ref. 3.7). The above accident sequences are dominant for both sets of hazard curves. In addition, the differences between the seismic risk estimates shown in Table 3.2 for the LLNL and the EPRI cases are due entirely to the differences between the two sets of hazard curves. That is, the system models, failure rates, and success logic were identical for both estimates.

The seismic hazard associated with the curves developed by EPRI was significantly less than that of the LLNL curves. Differences between these curves result primarily from differences between the methodology and assumptions used to develop the hazard curves. In the LLNL program, considerable emphasis was placed on a wide range of uncertainty in the ground-motion attenuation models, while a relatively coarse set of seismic tectonic provinces was used in characterizing each site. By contrast, in the EPRI program considerable emphasis was placed on a fine zonation for the tectonic provinces, and very little uncertainty in the ground-motion attenuation was considered. In any case, it is the difference between the two sets of hazard curves that causes the differences between the numeric estimates in Table 3.2.

2. Fire Accident Frequency Analysis

The fire-initiated accident frequency analyses performed for this report considered the impact of fires beginning in a variety of separate locations within the plant. Those locations found to be most important were:

- Emergency switchgear room,
- Control room,
- Auxiliary building, and
- Cable vault and tunnel.

In the emergency switchgear room, a fire is assumed to fail either control or power cables for both HPI and CCW, leading directly to a reactor coolant pump seal LOCA. No additional random failures were required for this sequence to lead to core damage. (Credit was given for operator recovery by crossconnecting the Unit 2 HPI system.) The identical scenario arises as the result of fires postulated in the auxiliary building and the cable vault and tunnel. Thus, fires in these three areas both cause the initiating event (a seal LOCA) and fail the system required to mitigate the scenario (i.e., HPI).

In the control room, a fire in a bench board was determined to lead to spurious actuation of a PORV with smoke-induced abandonment of the control room. A low probability of successful operator recovery actions from the remote shutdown panel (RSP) was assessed since the PORV closure status is not displayed at the RSP. In addition, the PORV block valve controls in the RSP are not routed independently of the control room bench board and thus may not function.

The frequency of fire-initiated accident scenarios in other locations contributed less than 10 percent to the total fire-initiated core damage frequency.

3.2.2 Important Plant Characteristics (Core Damage Frequency)

Characteristics of the Surry plant design and operation that have been found to be important in the analysis of core damage frequency include:

1. Crossties Between Units

The Surry plant has numerous crossties between similar systems at Units 1 and 2. Some of these were installed in order to comply with requirements of 10 CFR Part 50, Appendix R (fire protection) (Ref. 3.8) or high-energy line-break threats, and some were installed for operational reasons. Crossties exist for the auxiliary feedwater system, the charging pump system, the charging pump cooling system, and the refueling water storage tanks. These crossties are subject to technical specifications, their potential use is included in the plant operating procedures, and they are reviewed in operator training. The availability of such crossties was estimated to reduce the internal-event core damage frequency by approximately a factor of 3.

2. Diesel Generators

Surry is a two-unit site with three emergency diesel generators (DGs), one of which is a swing diesel (which can be aligned to one unit or the other), while many other PWR plants have dedicated diesels for each safety-grade power train (i.e., four DGs for a two-unit site). Each DG is self-cooled and supplied with a dedicated battery (independent of the batteries providing power to the vital dc buses) for starting. The latter two factors eliminate potential common-cause failure modes found important at other plants in this study (e.g., Peach Bottom and Grand Gulf). The Surry site also has a gas turbine generator. However, administrative procedures and design characteristics of support equipment (e.g., dc batteries and compressed air) preclude its use during a station blackout accident.

3. Reactor Coolant Pump Seals

At Surry, there are two diverse and independent methods for providing reactor coolant pump seal cooling: the component cooling water system and the charging system (which has its own dedicated cooling system). The only common support systems for seal cooling are ac and dc power. As such, reactor coolant pump seal LOCAs have been found important only in station blackout sequences. This is in contrast to some other PWR plants that have a dependency between charging pumps and the component cooling water system and thus greater potential for loss of seal cooling. Without cooling, the seals were expected to degrade or fail. The probability of seal failure upon loss of seal cooling was studied in detail by the expert panel elicitation (Ref. 3.9). Reflecting this, the Surry analyses have found that station blackout accident sequences with significant seal leakage are important contributors to the total frequency of core damage.

4. **Battery Capacity**

   For the Surry plant, the station Class 1E battery depletion time following station blackout has been estimated to be 4 hours (Ref. 3.5). The inability to ensure availability for longer times contributes significantly to the frequency of core damage resulting from station blackout accident sequences. The batteries are designed and tested for 2 hours. A 4-hour battery depletion time is considered realistic because of the margin in the design and possible load shedding.

5. **Capability for Feed and Bleed Core Cooling**

   In the Surry plant, the high-pressure injection system and the power-operated relief valves have the capability to provide feed and bleed core cooling in the event of loss of the cooling function of the steam generators. This capability to provide core cooling through feed and bleed is estimated to result in approximately a factor of 1.4 reduction in core damage frequency. Without the crossties of auxiliary feedwater to Unit 2, which enhances overall reliability of the auxiliary feedwater system, the benefit of feed and bleed cooling would be much greater.

3.2.3 Important Operator Actions

The estimation of accident sequence and total core damage frequencies depends substantially on the credit given to operating crews in performing actions before and during an accident. Failure to perform these actions correctly and reliably will have a substantial impact on estimated core damage frequency. For the Surry plant, actions found to be important are discussed below.

During loss of offsite power and station blackout, important actions required to be taken by the operating crew to prevent core damage include:

- Align alternative source of condensate to condensate storage tank

  The primary source of condensate for the AFW system is a 100,000-gallon tank. This is nominally sufficient for the duration of most station blackout events. But in the event that a steam generator becomes faulted, the increased AFW flow would require the provision of additional condensate water. This would involve manual local actions.

- Isolate condenser water box

  Surry has a somewhat unique gravity-fed service water system that relies on the head difference between the intake canal and the discharge canal to provide flow through service water heat exchangers. The intake canal is normally supplied with water by the circulating water pumps. These pumps are not provided with emergency power and are thus unavailable after a loss of offsite power. The condenser at each unit is provided with four inlet and four outlet isolation valves. These isolation valves are provided with emergency power. Each inlet isolation valve is provided with a hand wheel, located in the turbine building, in order to allow manual condenser isolation during station blackout to avoid draining the canal.

- Cool down and depressurize the RCS

  The Emergency Contingency Actions (ECAs) call for depressurization of the secondary side of the steam generators during a station blackout to provide cooldown and depressurization of the reactor coolant system. This action is done through manual, local valve lineups.

During steam generator tube rupture, the most important operator action is to cool down and depressurize the RCS within approximately 45 minutes after the event in order to prevent lifting the relief valves on the damaged steam generator. Other possible recovery actions considered in this accident sequence include: provision of an alternative source of steam generator feed flow in response to a loss of feed flow; crossconnect of HPI from Unit 2 or opening of alternative injection paths in response to failure of safety injection flow; and isolation of a damaged, faulted steam generator.

During small-break and medium-break LOCA accident sequences, two human actions are principally important in response to loss of core coolant injection or recirculation. These are:

- Cool down and depressurize the RCS

  RCS cooldown and depressurization is the procedure directed for all small-break LOCAs. This event is important to reduce the pressure in the RCS and thus reduce the leak rate. Successful cooldown and depressurization of the RCS will delay the need to go to recirculation cooling.

- Crossconnect high-pressure injection (HPI)

  In the event that HPI pumps or water sources are unavailable at Unit 1, HPI flow can be provided via a crosstie with the Unit 2 charging system. This crosstie requires an operator to locally open and/or close valves in the charging pump area. It was estimated that the crossconnect of HPI would require 15 to 20 minutes. This and other timing considerations were such that the HPI crossconnect was considered viable only for small and very small LOCAs.

3.2.4 Important Individual Events and Uncertainties (Core Damage Frequency)

As discussed in Chapter 2, the process of developing a probabilistic model of a nuclear power plant involves the combination of many individual events (initiators, hardware failures, operator errors, etc.) into accident sequences and eventually into an estimate of the total frequency of core damage. After development, such a model can also be used to assess the relative importance and contribution of the individual events. The detailed studies underlying this report have been analyzed using several event importance measures. The results of the analyses using two measures, "risk reduction" and "uncertainty" importance, are summarized below.

- Risk (core damage frequency) reduction importance measure (internal events)

  The risk-reduction importance measure is used to assess the change in core damage frequency as a result of setting the probability of an individual event to zero. Using this measure, the following individual events were found to cause the greatest reduction in the estimated core damage frequency if their probabilities were set to zero:

  - Loss of offsite power initiating event. The core damage frequency would be reduced by approximately 61 percent.

  - Failure of diesel generator number one to start. The core damage frequency would be reduced by approximately 25 percent.

  - Probability of not recovering ac electric power between 3 and 7 hours after loss of offsite power. The core damage frequency would be reduced by approximately 24 percent.

  - Failure to recover diesel generators. The core damage frequency would be reduced by approximately 18 to 21 percent.

- Uncertainty importance measure (internal events)

  A second importance measure used to evaluate the core damage frequency results is the uncertainty importance measure. For this measure, the relative contribution of the uncertainty of groups of component failures and basic events to the uncertainty in total core damage frequency is calculated. Using this measure, the following event groups were found to be most important:

  - Probabilities of diesel generators failing to start when required;

  - Probabilities of diesel generators failing to run for 6 hours;

  - Frequency of loss of offsite power; and

  - Frequency of interfacing-system LOCA.

It should be noted that many events each contribute a small amount to the uncertainty in core damage frequency; no single event dominates the uncertainty.

3.3 Containment Performance Analysis

3.3.1 Results of Containment Performance Analysis

The Surry containment system uses a subatmospheric concept in which the containment building housing the reactor vessel, reactor coolant system, and secondary system's steam generator is maintained at 10 psia. The containment building is a reinforced concrete structure with a volume of 1.8 million cubic feet. Its design basis pressure is 45 psig, whereas its mean failure pressure is estimated to be 126 psig. As previously discussed in Chapter 2, the method used to estimate accident loads and containment structural response for Surry made extensive use of expert judgment to interpret and supplement the limited data available.

The potential for early Surry containment failure is of major interest in this risk analysis. The principal threats identified in the Surry risk analyses (Ref. 3.2) as potentially leading to early containment failure are: (1) pressure loads, i.e., hydrogen combustion and direct containment heating due to ejection of molten core material via the rapid expulsion of hot steam and gases from the reactor coolant system; and (2) in-vessel steam explosions leading to vessel failure with the vessel upper head being ejected and impacting the containment building dome area (the so-called alpha-mode failure). Containment bypass (such as failures of reactor coolant system isolation check valves in the emergency core cooling system or steam generator tubes) is another serious threat to the integrity of the containment system.

The results of the Surry containment analysis are summarized in Figures 3.5 and 3.6. Figure 3.5 displays information in which the conditional probabilities of seven containment-related accident progression bins; e.g., VB, alpha, early CF, are presented for each of seven plant damage states; e.g., loss of offsite power. This information indicates that, on a plant damage state frequency-weighted average,* the conditional mean probability from internally initiated accidents of: (1) early containment failure is about 0.01, (2) late containment failure (basemat melt-through or leakage) is about 0.06, (3) direct bypass of the containment is about 0.12, and (4) no containment failure is 0.81. Figure 3.6 further displays the conditional probability distribution of early containment failure for each plant damage state to show the estimated range of uncertainties in these containment failure predictions. The important conclusions to be drawn from the information in Figures 3.5 and 3.6 are: (1) the mean conditional probability of early containment failure from internal events is low; i.e., less than 0.01; (2) the principal containment release mechanism is bypass due to interfacing-system LOCA; and (3) external initiating events such as fire and earthquakes produce higher early and late containment failure probabilities.

*Each value in the column in Figure 3.5 labeled "All" is obtained by calculating the products of individual accident progression bin conditional probabilities for each plant damage state and the ratio of the frequency of that plant damage state to the total core damage frequency.

The accident progression analyses performed for this report are particularly noteworthy in that, for core melt accidents at Surry, there is a high probability that the reactor coolant system (RCS) will be at relatively low pressures (less than 200 psi) at the time of molten core penetration of the lower reactor vessel head, thereby reducing the potential for direct containment heating (DCH). There are several reasons for concluding that the RCS will be at low system pressure such as: stuck-open PORVs, operator depressurization, failed reactor coolant pump seals, induced failures of RCS piping due to high temperatures, and the relative "mix" of plant damage states (i.e., for the frequency of plant damage states initially at high versus low RCS pressures). Accordingly, it has been concluded that the potential for early containment failure due to the phenomenon of DCH is less in the risk analyses underlying this report relative to previous studies (Ref. 3.10) on the basis of a combination of higher probabilities of low RCS pressures (discussed above), lower calculated pressures given direct containment heating, and greater estimated strength of the Surry containment building (Ref. 3.2). (See Section C.5 of Appendix C for additional discussion of DCH and why its importance is now less.)

Additional discussions on containment performance (for all studied plants) are provided in Chapter 9.

3.3.2 Important Plant Characteristics (Containment Performance)

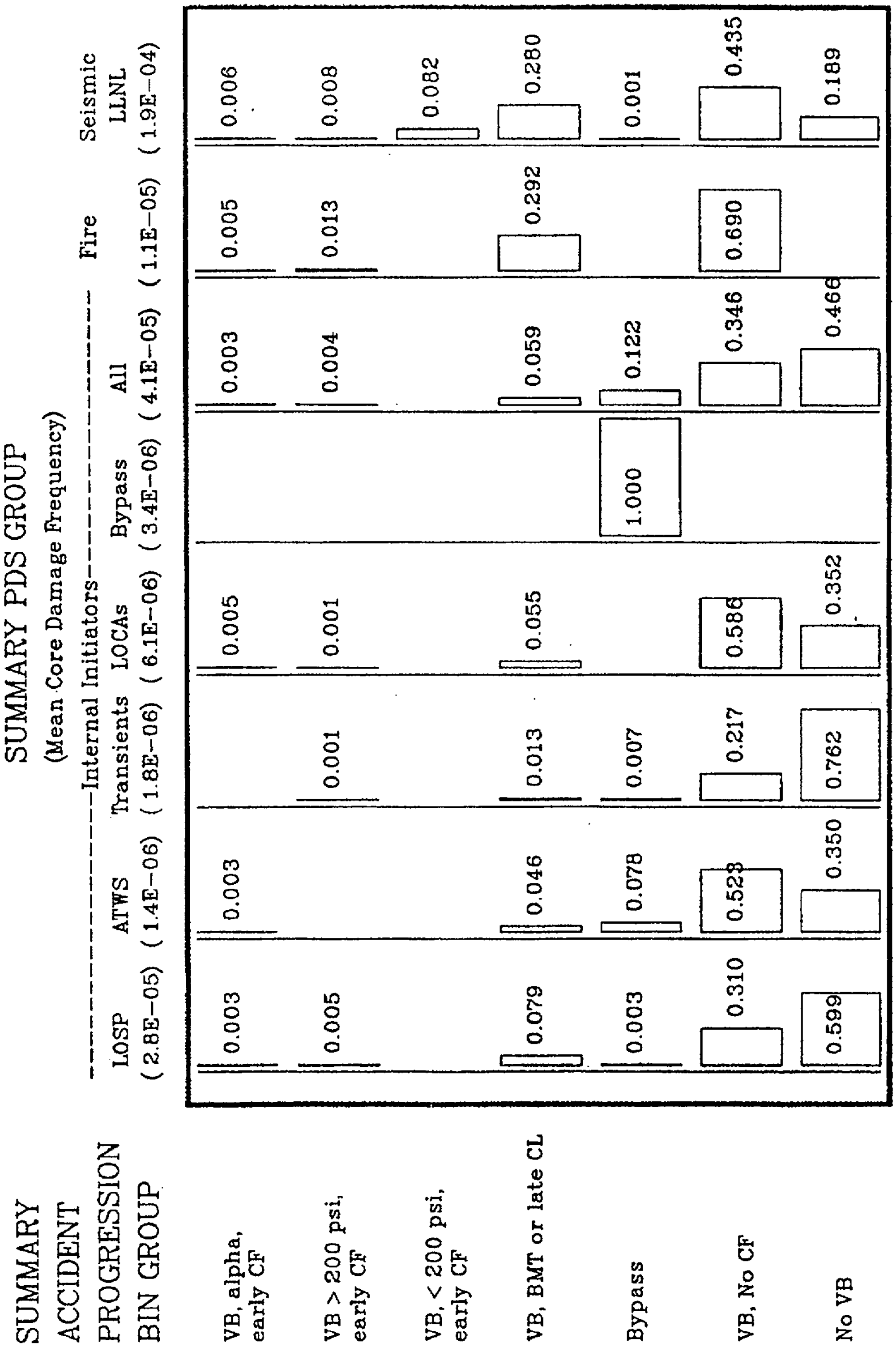
Characteristics of the Surry plant design and operation that are unique to the containment building during core damage accidents include:

1. **Subatmospheric Containment Operation**

   The Surry containment is maintained at a subatmospheric pressure (10 psia) during operation with a continual monitoring of the containment leakage. As a result, the likelihood of pre-existing leaks of significant size is negligible.

2. **Post-Accident Heat Removal System**

   The Surry containment does not have fan cooler units that are qualified for post-accident heat removal as do some other PWR plants. Containment (and core) heat removal



Key: BMT = Basemat Melt-Through
CF = Containment Failure
CL = Containment Leak
VB = Vessel Breach

Figure 3.5 Conditional probability of accident progression bins at Surry.

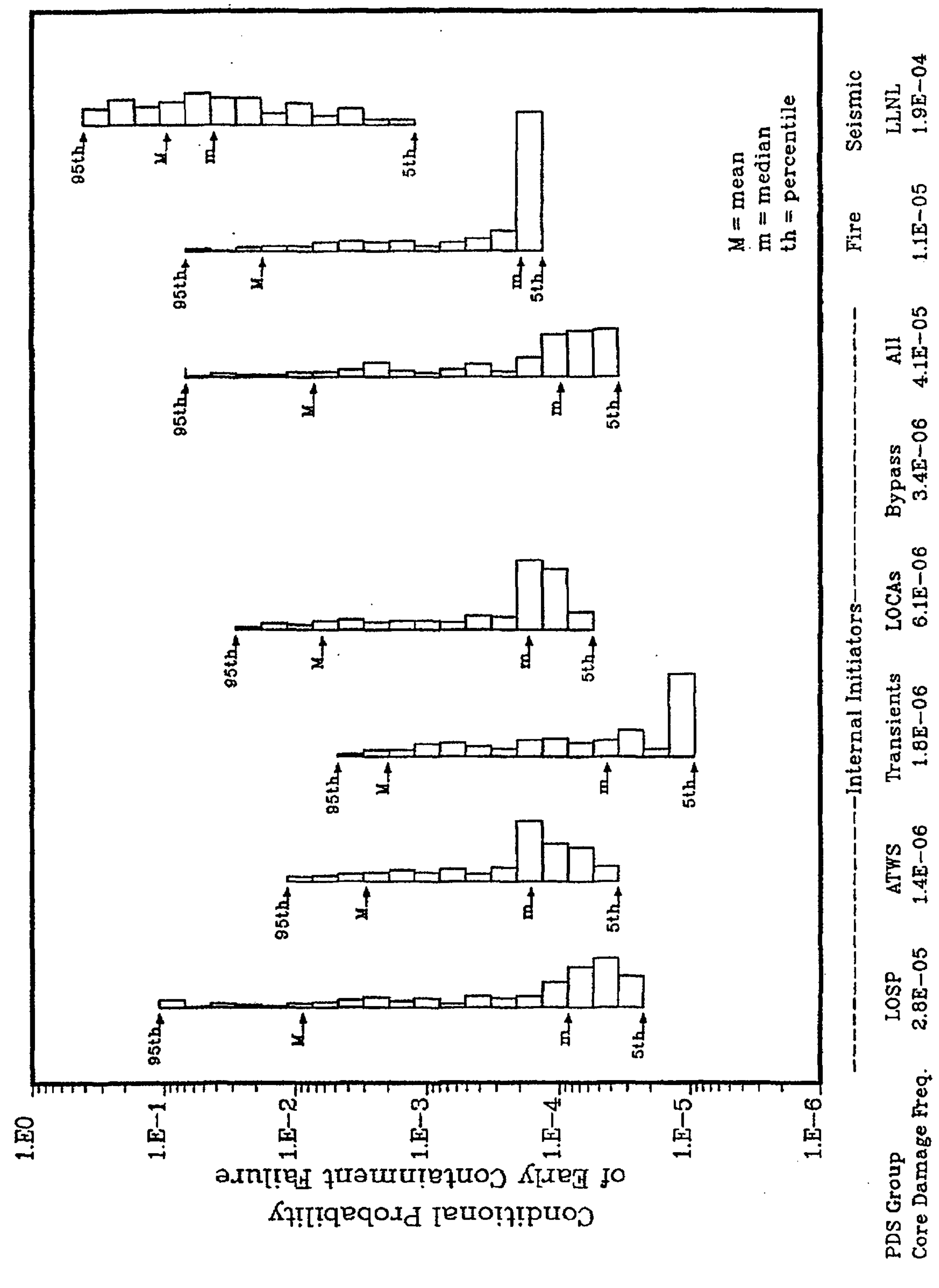


Figure 3.6 Conditional probability distributions for early containment failure at Surry.

following an accident is provided by the containment spray recirculation system, whereas, in some PWR plants, post-accident heat removal can also be provided by the residual heat removal system heat exchangers in the emergency core cooling system.

3. **Reactor Cavity Design**

   The reactor cavity area is not connected directly with the containment sump area. As a result, if the containment spray systems fail to operate during an accident, the reactor cavity will be relatively dry. The amount of water in the cavity can have a significant influence on phenomena that can occur after reactor vessel lower head failure, such as magnitude of containment pressurization from direct containment heating and post-vessel failure steam generation, the formation of coolable debris beds, and the retention of radioactive material released during core-concrete interactions.

4. **Containment Building Design**

   The containment volume and high failure pressure provide considerable capacity for accommodation of severe accident pressure loads.

3.4 Source Term Analysis

3.4.1 Results of Source Term Analysis

In the Surry plant, the absolute frequency of an early failure of the containment* due to the loads produced in a severe accident is small. Although the absolute frequency of containment bypass is also small, for internal accident initiators it is greater than the absolute early failure frequency. Thus, bypass sequences are the more likely means of obtaining a large release of radioactive material. Figure 3.7 illustrates the distribution of source terms associated with the accident progression bin representing containment bypass. The range of release fractions is quite large, primarily as the result of the range of parameters provided by the experts. The magnitude of the release for many of the elemental groups is also large, indicative of a potentially serious accident. Typically, consequence analysis codes only predict the occurrence of early fatalities in the surrounding population when the release fractions of the volatile groups (iodine, cesium, and tellurium) exceed approximately 10 percent (Ref. 3.11). For the bypass accident progression bin, the median value for the volatile radionuclides is approximately at the 10 percent level whereas for the early containment failure bin not shown, the releases are lower. The median values are somewhat smaller than 10 percent, but the ranges extend to approximately 30 percent.

In contrast to the large source term for the bypass bin, Figure 3.8 provides the range of source terms predicted for an accident progression bin involving late failure of the containment. The fractional release of radionuclides for this bin is several orders of magnitude smaller than for the bypass bin, except for iodine, which can be reevolved late in the accident. It should be noted that, for many of the elemental groups, the mean of the distribution falls above the 95th percentile value. For distributions that occur over a range of many orders of magnitude, sampling from the extreme tail of the distribution (at the high end) can dominate and cause this result.

Additional discussion on source term perspectives is provided in Chapter 10.

3.4.2 Important Plant Characteristics (Source Term)

Plant design features that affect the mode and likelihood of containment failure also influence the magnitude of the source term. These features were described in the previous section. Plant features that have a more direct influence on the source term are described in the following paragraphs.

1. **Containment Spray System**

   The Surry plant has an injection spray system that uses the refueling water storage tank as a water source and a recirculation spray system that recirculates water from the containment sump. Sprays are an effective means for removing airborne radioactive aerosols. For sequences in which sprays operate throughout the accident, it is most likely that the containment will not fail and the leakage to the environment will be minor. If the containment does fail late in the accident following extended spray operation, analyses indicate that the release of aerosols will be extremely small. Even in a station blackout case with delayed recovery of sprays, condensation of steam from the air, and a subsequent hydrogen explosion that fails containment, Source Term Code Package (STCP) analyses indicate that spray operation results in substantially reduced source terms (Ref. 3.12).

*In this section, the absolute frequencies of early containment failure are discussed (i.e., including the frequencies of the plant damage states). This is in contrast to the previous section, which discusses conditional failure probabilities (i.e., given that a plant damage state occurs).

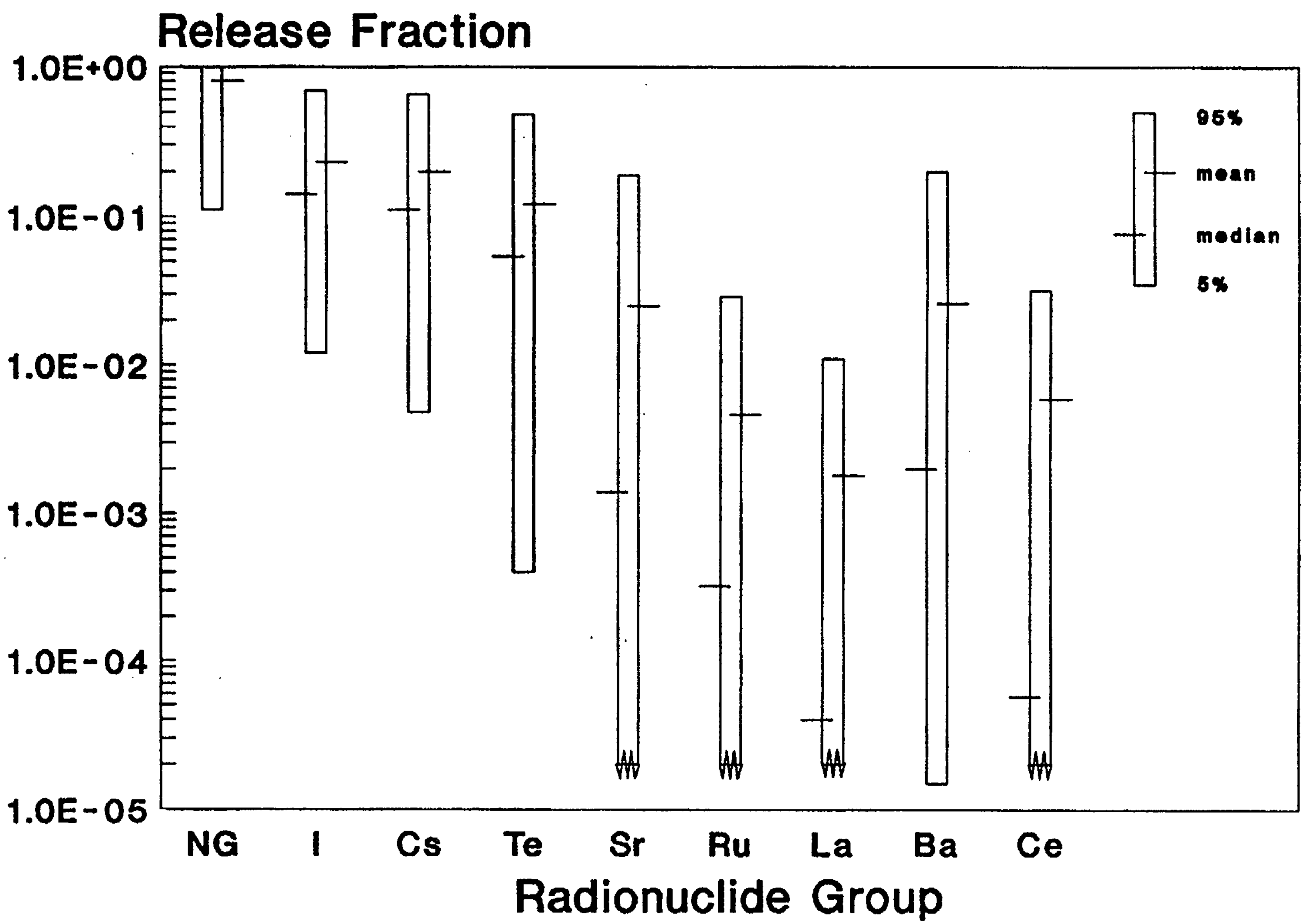


Figure 3.7 Source term distributions for containment bypass at Surry.

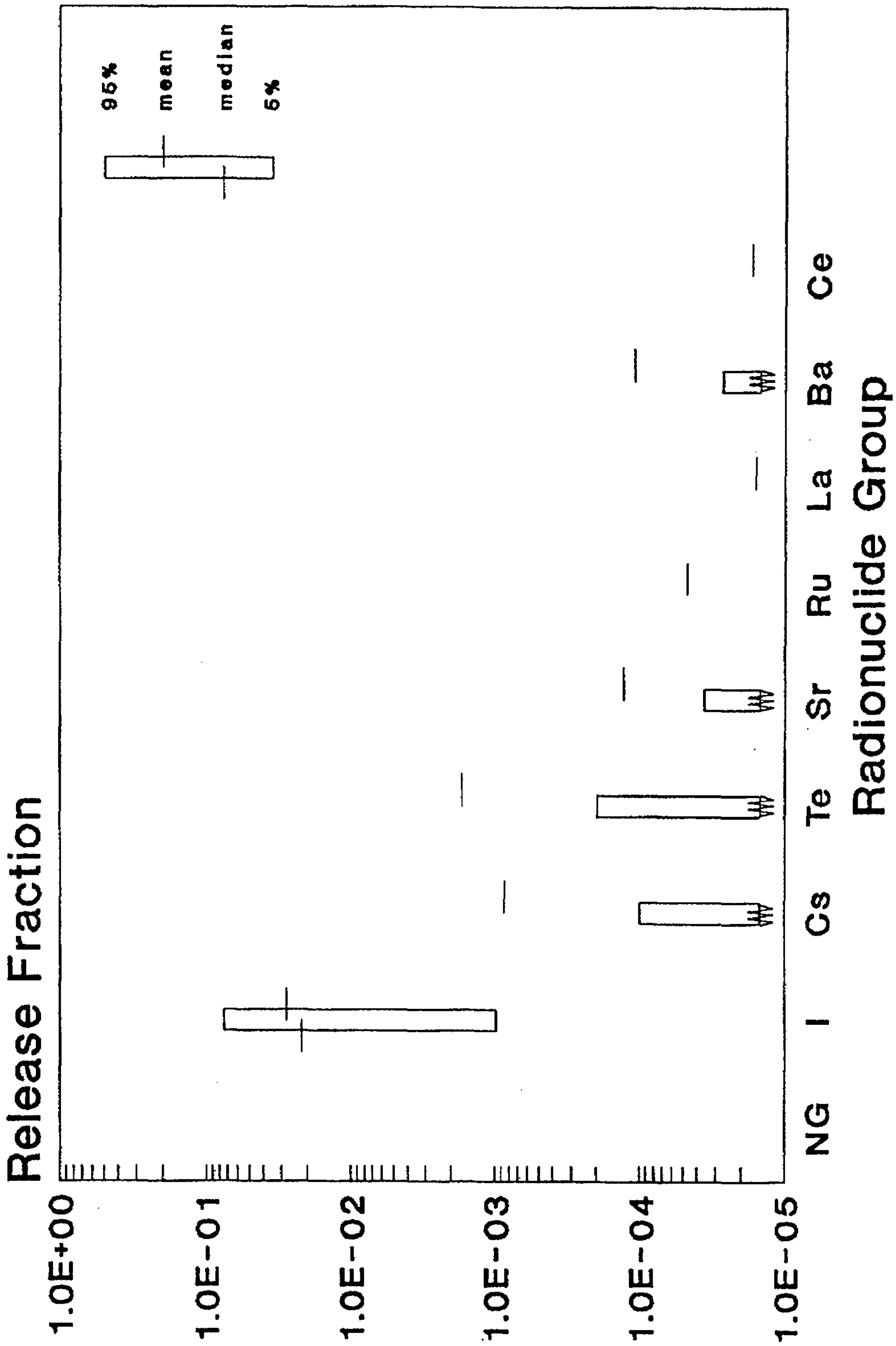


Figure 3.8 Source term distributions for late containment failure at Surry.

Sprays are not always effective in reducing the source term, however. The risk-dominant containment bypass sequences are largely unaffected by operation of the spray systems. Early containment failure scenarios involving high-pressure melt ejection have a component of the release that occurs almost simultaneously with containment failure, for which the sprays would not be effective.

In addition to removing aerosols from the atmosphere, containment sprays are an important source of water to the reactor cavity at Surry, which is otherwise dry. A coolable debris bed can be established in the cavity, preventing interactions between the hot core and concrete. If a coolable debris bed is not formed, a pool of water overlaying the hot core as it attacks concrete can effectively mitigate the release of radioactive material to the containment from this interaction.

2. **Cavity Configuration**

Water collecting on the floor of the Surry containment cannot flow into the reactor cavity. As a result, the cavity will be dry at the time of vessel meltthrough unless the containment spray system has operated. As discussed earlier, water in the cavity can have a substantial effect on mitigating or eliminating the release of radioactive material from the molten core-concrete interaction.

3.5 Offsite Consequence Results

Figures 3.9 and 3.10 display the frequency distributions in the form of graphical plots of complementary cumulative distribution functions (CCDFs) of four offsite consequence measures—early fatalities, latent cancer fatalities, and the 50-mile and entire site region population exposures (in person-rems). The CCDFs in Figures 3.9 and 3.10 include contributions from all source terms associated with reactor accidents caused by the internal initiating events and fire, respectively. Four CCDFs, namely, the 5th percentile, 50th percentile (median), 95th percentile, and the mean CCDFs, are shown for each consequence measure.

Surry plant-specific and site-specific parameters were used in the consequence analysis for these CCDFs. The plant-specific parameters included source terms and their frequencies, the licensed thermal power (2441 MWt) of the reactor, and the approximate physical dimensions of the power plant building complex. The site-specific parameters included exclusion area radius (520 meters), meteorological data for 1 full year collected at the site meteorological tower, the site region population distribution based on the 1980 census data, topography (fraction of the area that is land—the remaining fraction is assumed to be water), land use, agricultural practice and productivity, and other economic data for up to 1,000 miles from the Surry plant.

The consequence estimates displayed in these figures have incorporated the benefits of the following protective measures: (1) evacuation of 99.5 percent of the population within the 10-mile plume exposure pathway emergency planning zone (EPZ), (2) early relocation of the remaining population only from the heavily contaminated areas both within and outside the 10-mile EPZ, and (3) decontamination, temporary interdiction, or condemnation of land, property, and foods contaminated above acceptable levels.

The population density within the Surry 10-mile EPZ is about 230 persons per square mile. The average delay time before evacuation (after a warning prior to radionuclide release) from the 10-mile EPZ and average effective evacuation speed used in the analyses were derived from information contained in a utility-sponsored Surry evacuation time estimate study (Ref. 3.13) and the NRC requirements for emergency planning.

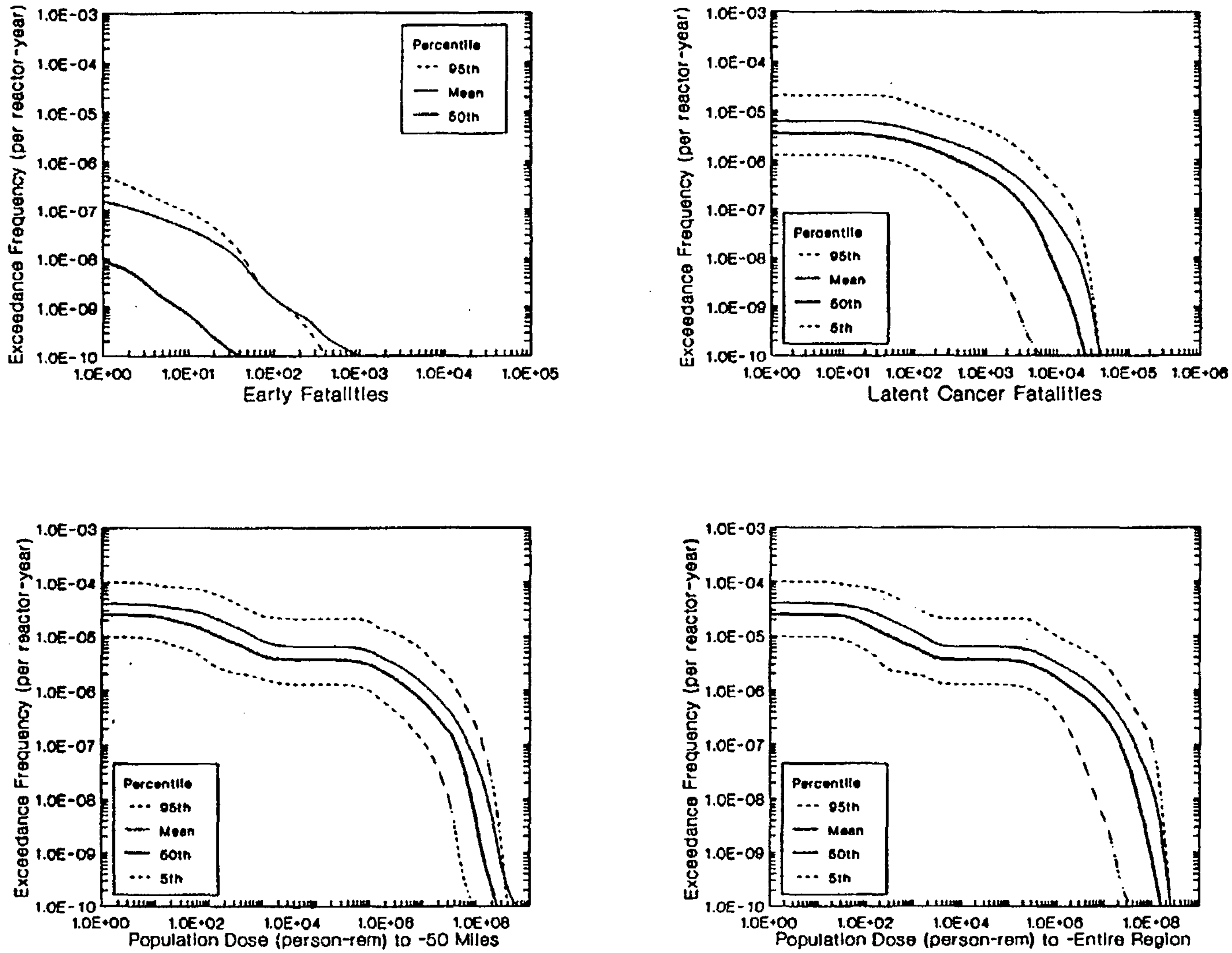
The results displayed in Figures 3.9 and 3.10 are discussed in Chapter 11.

3.6 Public Risk Estimates

3.6.1 Results of Public Risk Estimates

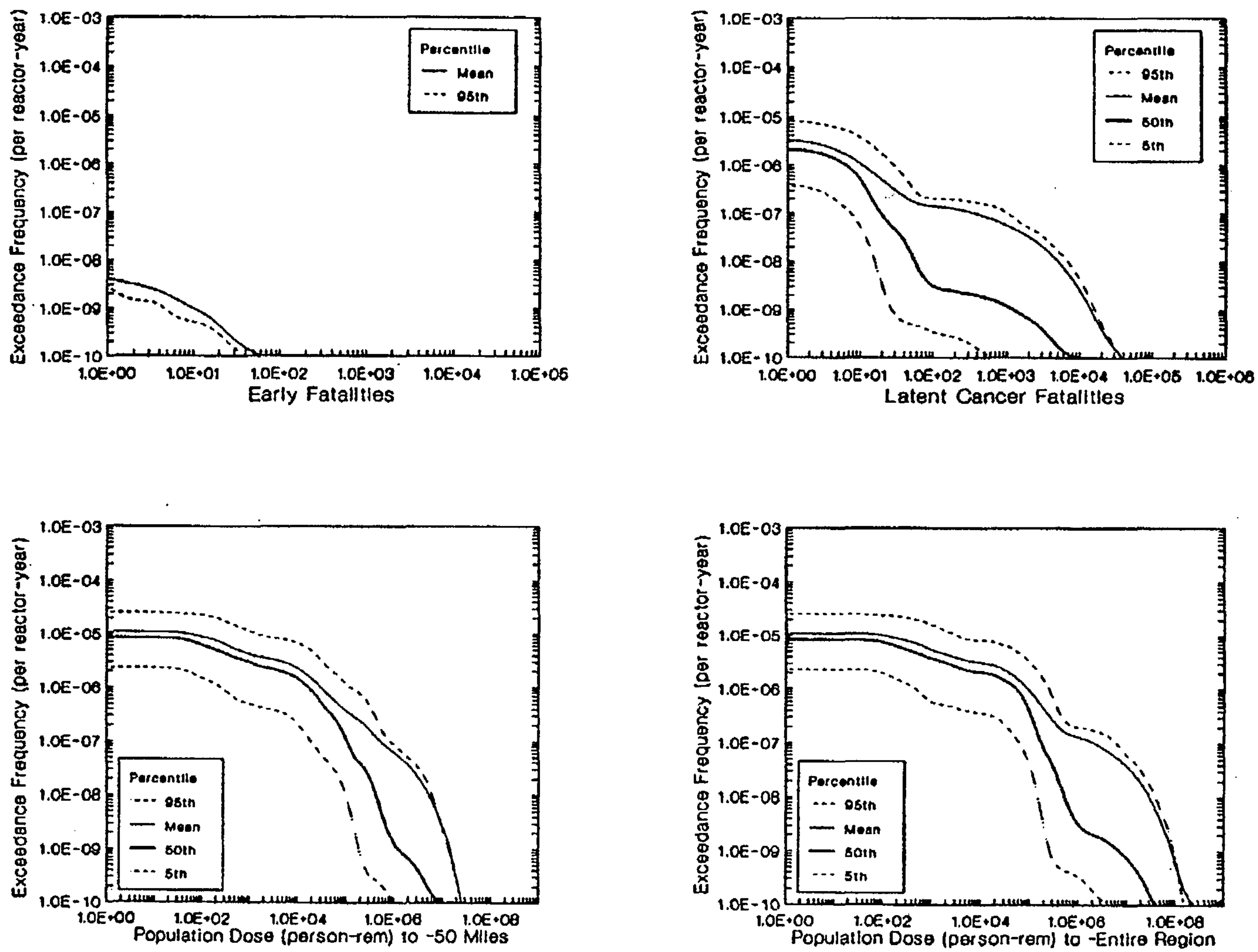
A detailed description of the results of the Surry risk analysis is provided in Reference 3.2. For this summary report, results are provided for the following measures of public risk:

- Early fatality risk,
- Latent cancer fatality risk,
- Population dose within 50 miles of the site,
- Population dose within the entire site region,
- Individual early fatality risk in the population within 1 mile of the Surry exclusion area boundary, and
- Individual latent cancer fatality risk in the population within 10 miles of the Surry site.



Note: As discussed in Reference 3.4, consequences at frequencies estimated 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 3.9 Frequency distributions of offsite consequence measures at Surry (internal initiators).



Note: As discussed in Reference 3.4, consequences at frequencies estimated 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 3.10 Frequency distributions of offsite consequence measures at Surry (fire initiators).

The first four of the above measures are commonly used measures in nuclear power plant risk studies. The last two are those used to compare with the NRC safety goals (Ref. 3.14).

3.6.1.1 Internally Initiated Accident Sequences

The results of the risk studies using the above measures are provided in Figures 3.11 through 3.13 for internally initiated accidents. The figures display the variabilities in mean risks estimated from the meteorology-averaged conditional mean values of the consequence measures. For the first two measures, the results of the first risk study of Surry, the Reactor Safety Study (Ref. 3.3), are also provided. As may be seen, both the early fatality risks and latent cancer fatality risks are lower than those of the Reactor Safety Study. The early fatality risk distribution, however, has a longer tail at the low end indicating a belief by the experts that there is a finite probability that risks may be orders of magnitude lower than those of the Reactor Safety Study. The risks of population dose within 50 miles of the plant site as well as within the entire site region are very low. Individual early fatality and latent cancer fatality risks are well below the NRC safety goals.

For the early and latent cancer fatality risk measures, the Reactor Safety Study values lie in the upper portions of the present risk range. This is because of the current estimates of better containment performance and source terms. The estimated probability of early containment failure in this study is significantly lower than the Reactor Safety Study values. The source term ranges of the Reactor Safety Study are comparable with the upper portions of the present study. The median core damage frequencies of the two studies, however, are about the same (2.3E–5 per reactor year for this study compared to 4.6E–5 per reactor year for the Reactor Safety Study). A more detailed comparison between results is provided in Chapters 12.

The risk results shown in Figure 3.11 have been analyzed to determine the relative contributions of plant damage states and containment-related accident progression bins to mean risk. The results of this analysis are provided in Figures 3.14 and 3.15. As may be seen, the mean early and latent cancer fatality risks of the Surry plant are principally due to accidents that bypass the containment building (interfacing-system LOCA (Event V) and steam generator tube ruptures). Details of these accident sequences are provided in Section 3.2.1.1. It should be noted from these discussions that for the steam generator tube rupture accident, if corrective or protective actions are taken (e.g., alternative sources of water are made available, emergency response is initiated*) before the refueling water storage tank water is totally depleted, i.e., within about a 10-hour period after start of the accident, risks from this accident may be substantially reduced.

3.6.1.2 Externally Initiated Accident Sequences

The Surry plant has been analyzed for two externally initiated accidents: earthquakes and fire (see Section 3.2.1.2). The fire risk analysis has been performed, including estimates of consequences and risk, while the seismic analysis has been conducted up to the containment performance (as discussed in Chapter 2). Sensitivity analyses of seismic risk at Surry are provided in Reference 3.2.

Results of fire risk analysis (variabilities in mean risks estimated from meteorology-averaged conditional mean values of the consequence measures) of Surry are shown in Figures 3.16 through 3.18 for the early fatality, latent cancer fatality, population dose (within 50 miles of the site and within the entire site region), and individual early and latent cancer fatality risks. As can be seen, the risks from fire are substantially lower than those from internally initiated events.

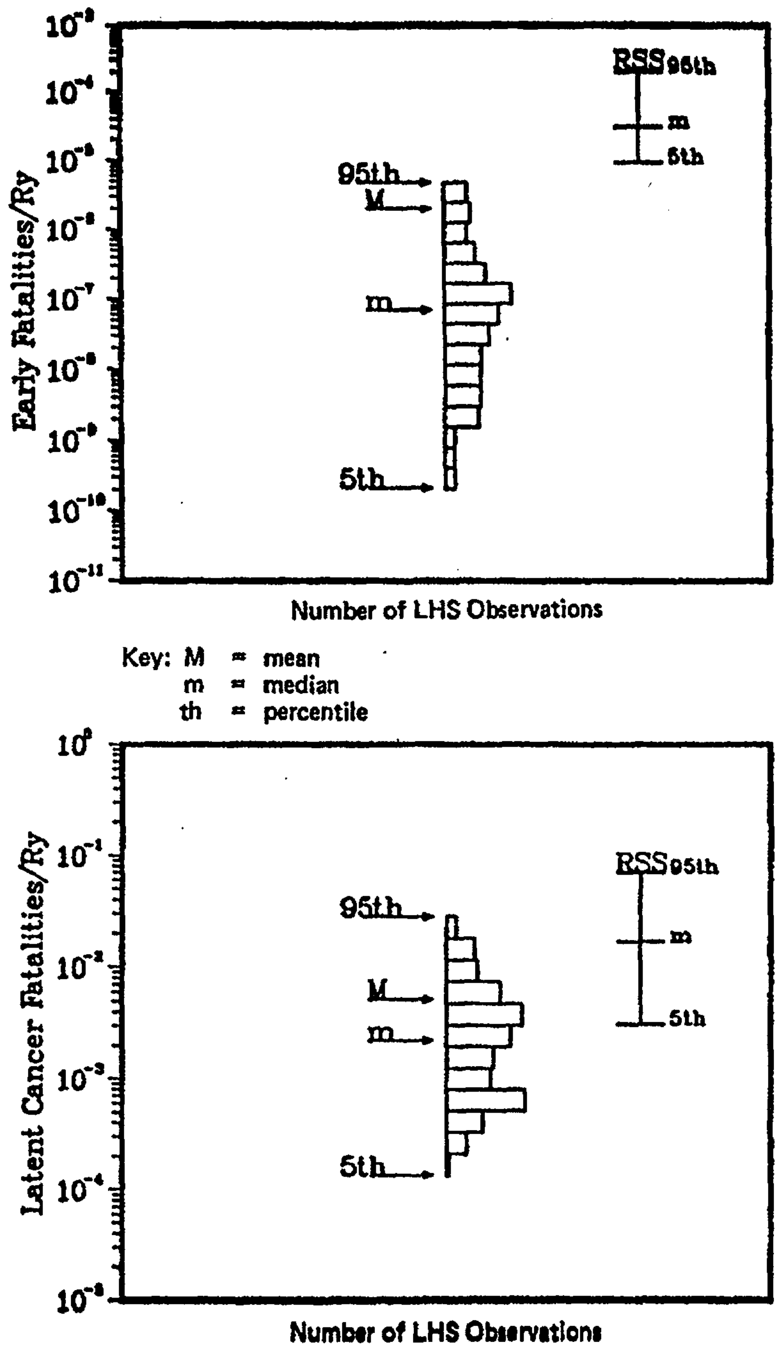
Major contributors to early and latent cancer fatality risks are shown in Figure 3.19. (Note that there are no bypass initiating events in the fire plant damage state.) The most risk-important sequence is a fire in the emergency switchgear room that leads to loss of ac power throughout the station. The principal risk-important accident progression bin is early containment failure with the reactor coolant system at high pressure (>200 psia) at vessel breach leading to direct containment heating.

Additional discussion of risk perspectives (for all five plants studied) is provided in Chapter 12.

3.6.2 Important Plant Characteristics (Risk)

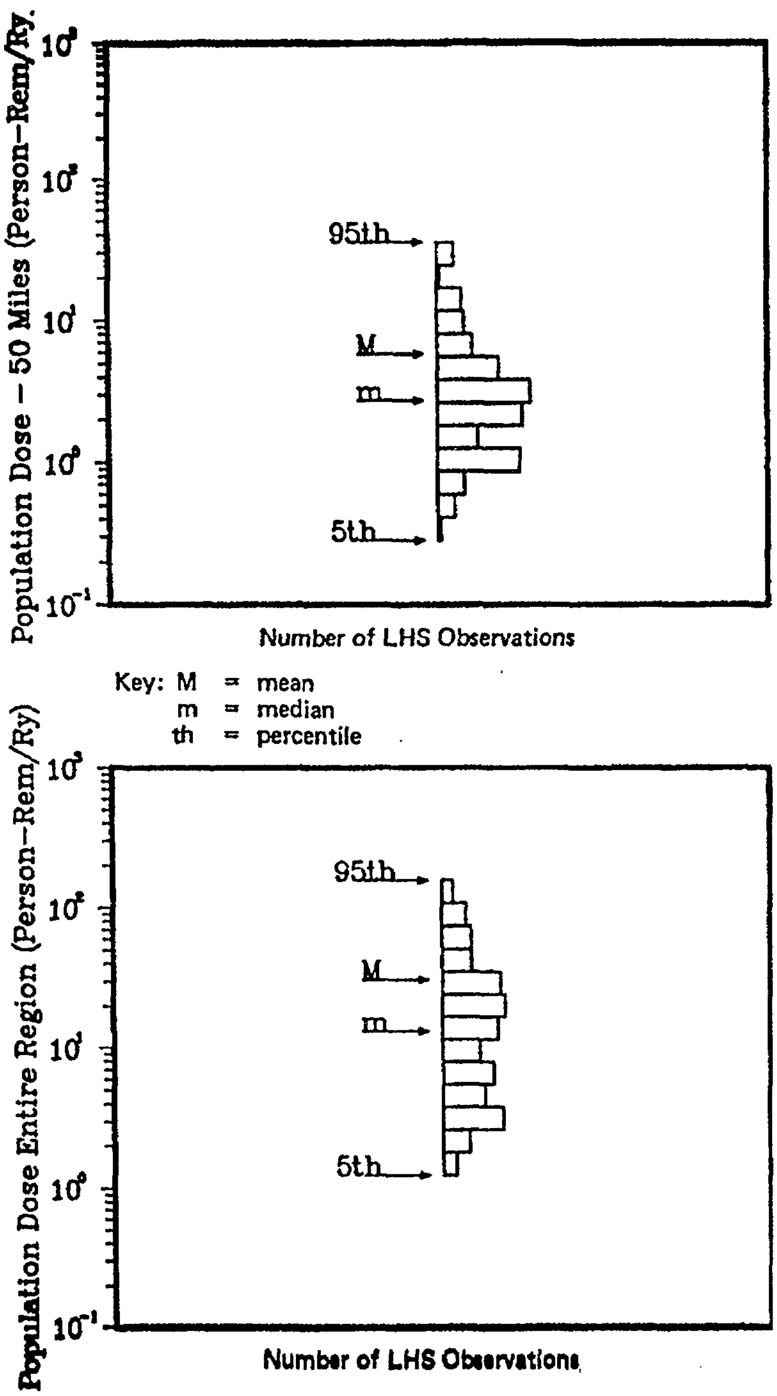
The plant characteristics discussed in Section 3.2.2 that were important in the analysis of core damage frequency were primarily related to the station blackout accident sequences and have not been found to be important in the risk analysis.

*See Chapter 11 for sensitivity of offsite consequences to alternative modes of emergency response.



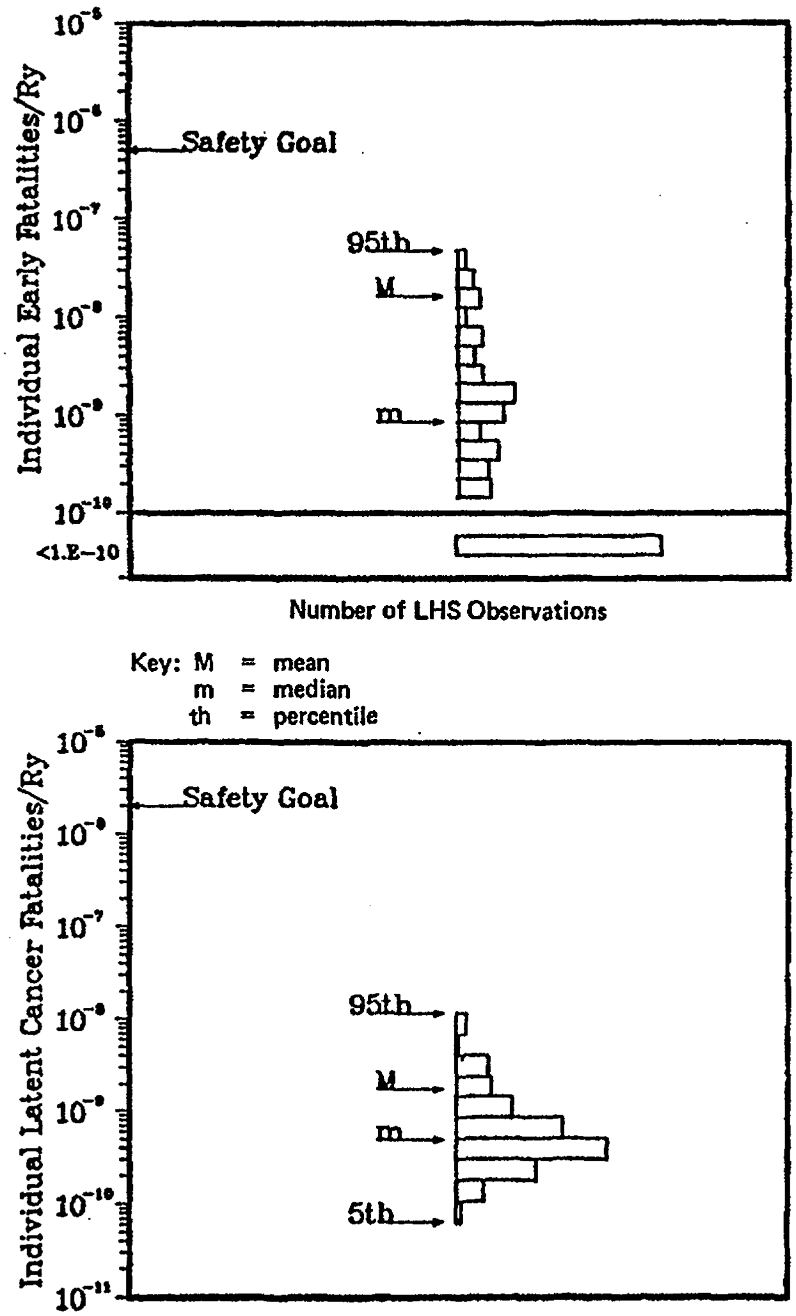
Note: As discussed in Reference 3.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 3.11 Early and latent cancer fatality risks at Surry (internal initiators).



Note: As discussed in Reference 3.4, 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 3.12 Population dose risks at Surry (internal initiators).



Note: As discussed in Reference 3.4, 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 3.13 Individual early and latent cancer fatality risks at Surry (internal initiators).

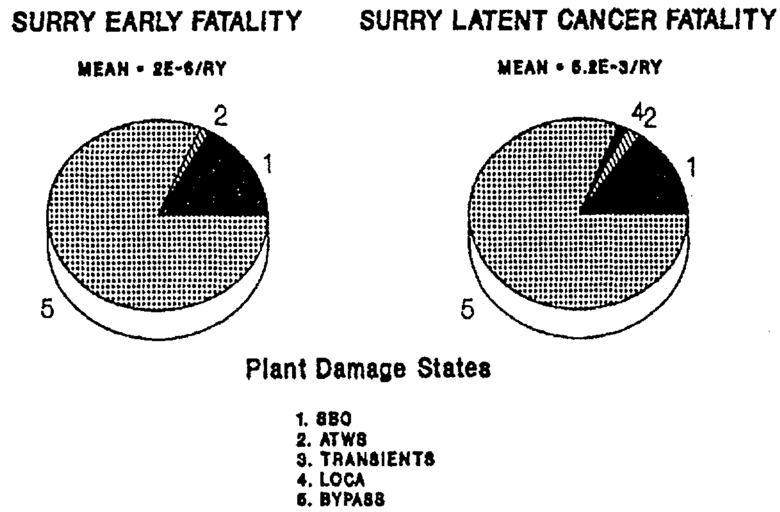


Figure 3.14 Major contributors (plant damage states) to mean early and latent cancer fatality risks at Surry (internal initiators).

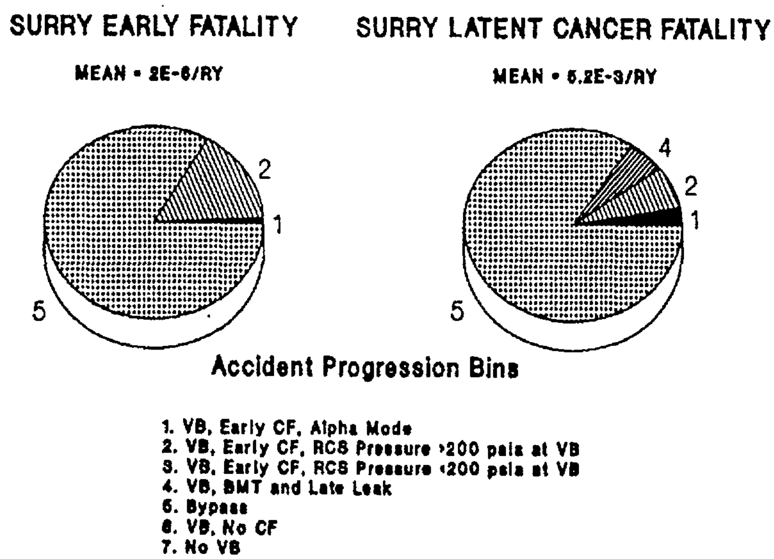
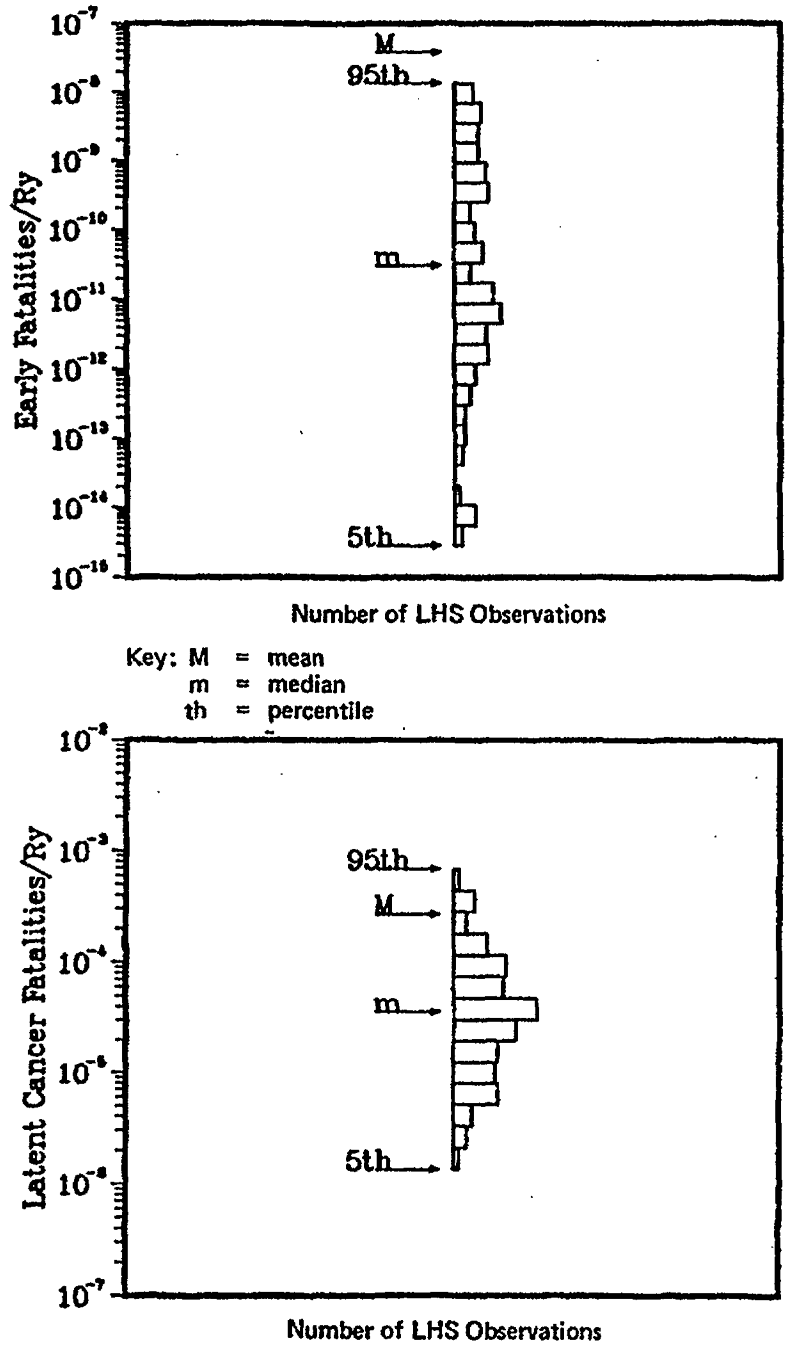
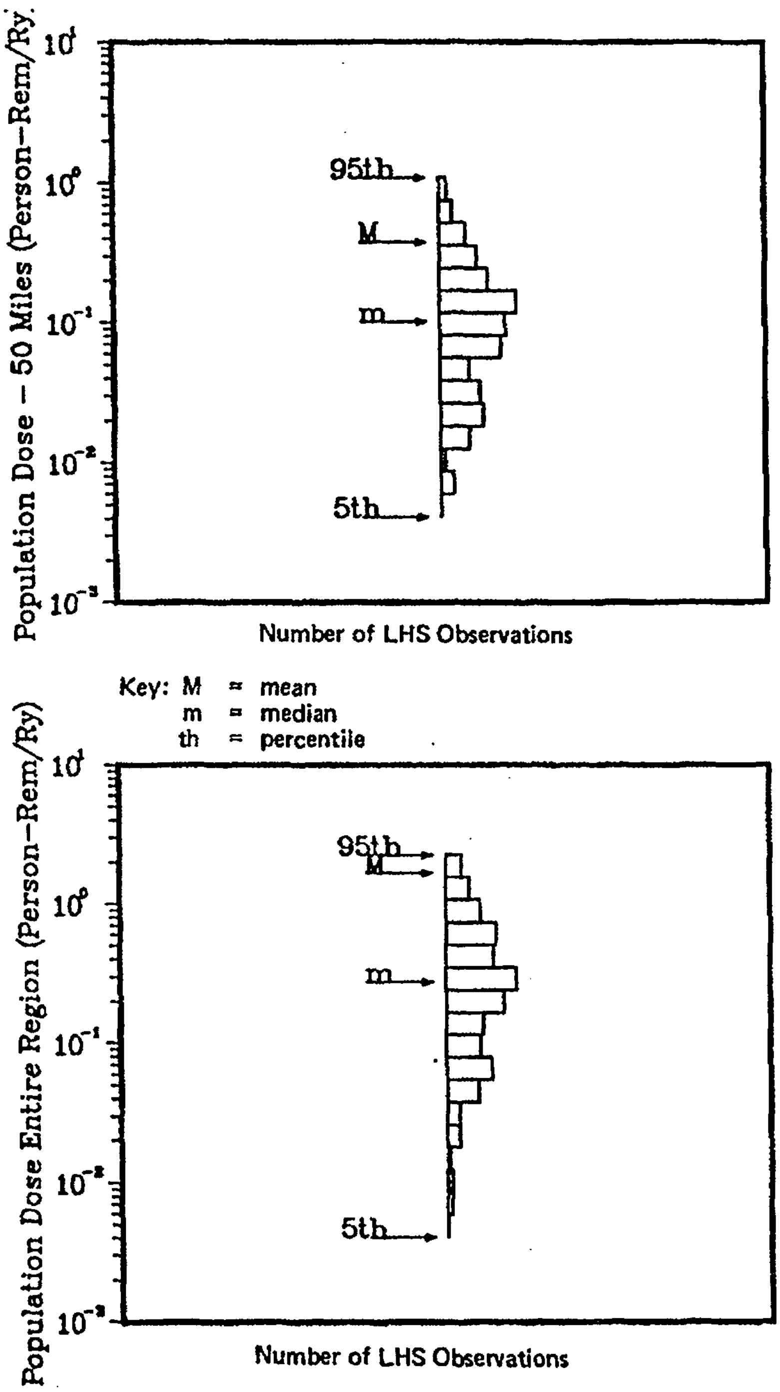


Figure 3.15 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Surry (internal initiators).



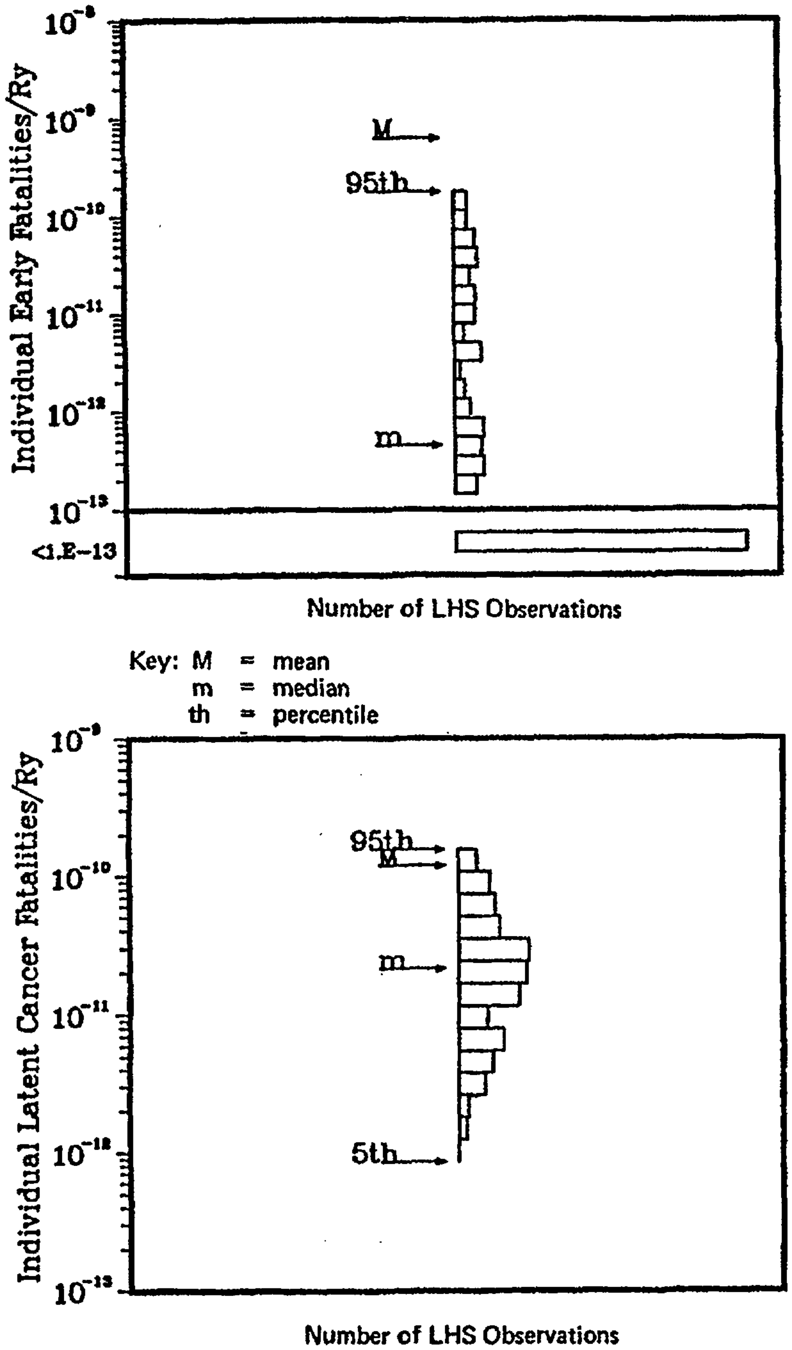
Note: As discussed in Reference 3.4, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 3.16 Early and latent cancer fatality risks at Surry (fire initiators).



Note: As discussed in Reference 3.4, 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 3.17 Population dose risks at Surry (fire initiators).



Note: As discussed in Reference 3.4, 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 3.18 Individual early and latent cancer fatality risks at Surry (fire initiators).

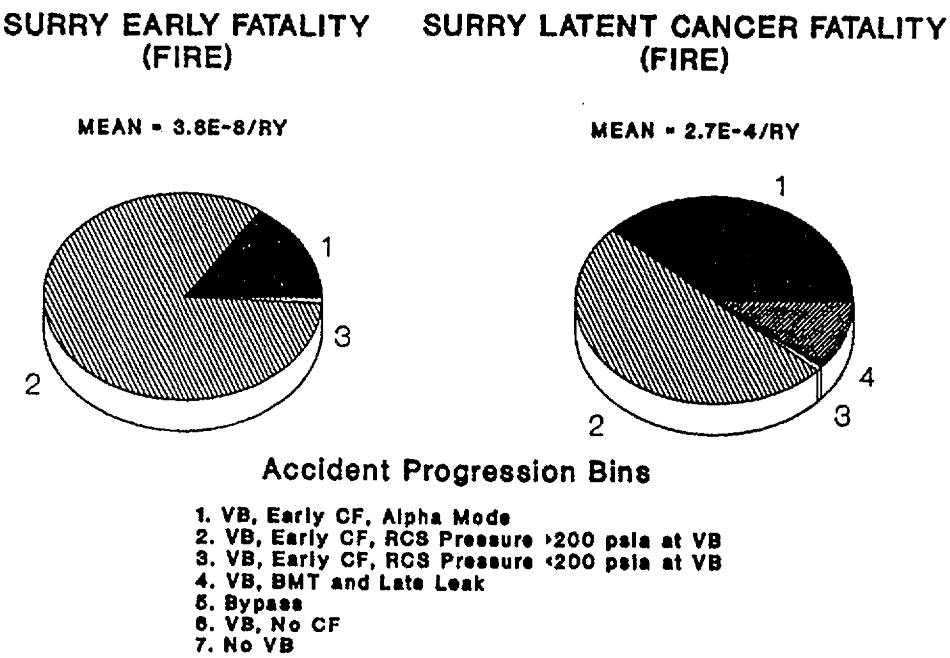


Figure 3.19 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Surry (fire initiators).

That is, because of the high consequences of the containment bypass sequences and low frequency of early containment failures, Event V and SGTR were more important risk contributors in the Surry analysis. The following general observations can be made from the risk results:

- The Surry containment appears robust, with a low conditional probability of failure (early or late). This is responsible, to a large extent, for the low risk estimates for the Surry plant. (In comparison with other plants studied in this report, risks for Surry are relatively high; but, in the absolute sense, these risks are very low and are well below NRC safety goals, as can be seen in Chapter 12.)

- Early fatality risk is dominated by bypass accidents, primarily from an interfacing-system LOCA. This accident leads to rapid core damage; the radioactive release is assessed to take place before evacuation is complete. Steam generator tube rupture accident sequences with stuck-open SRVs result in very late core melt; evacuation is assessed to be complete before the release is estimated to occur.

- The configuration of low-pressure piping outside the containment leads to a high probability that the release from an interfacing-system LOCA would be partially scrubbed by overlaying water. If the release were to take place without such scrubbing, the contribution to early fatality risk would be higher.

- Depressurization of the reactor coolant system by deliberate or inadvertent means plays an important role in the progression of severe accidents at Surry in that it decreases the probability of containment failure by high-pressure melt ejection and direct containment heating.

- Risks from accidents initiated by fires are dominated by early containment failures and are estimated to be much lower than those from internally initiated accidents.

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4. PEACH BOTTOM PLANT RESULTS

4.1 Summary Design Information

The Peach Bottom Atomic Power Station is a General Electric boiling water reactor (BWR-4) unit of 1065 MWe capacity housed in a Mark I containment constructed by Bechtel Corporation. Peach Bottom Unit 2, analyzed in this study, began commercial operation in July 1974 under the operation of Philadelphia Electric Company (PECo). Some important system design features of the Peach Bottom plant are described in Table 4.1. A general plant schematic is provided in Figure 4.1.

This chapter provides a summary of the results obtained in the detailed risk analyses underlying this report (Refs. 4.1 and 4.2). A discussion of perspectives with respect to these results is provided in Chapters 8 through 12.

4.2 Core Damage Frequency Estimates

4.2.1 Summary of Core Damage Frequency Estimates

The core damage frequency and risk analyses performed for this study considered accidents initiated by both internal and external events (Refs. 4.1 and 4.2). The core damage frequency results obtained from internal events are displayed in graphical form as a histogram in Figure 4.2 (Section 2.2.2 discusses histogram development). The core damage frequency results obtained from internal and external events are provided in tabular form in Table 4.2.

The Peach Bottom plant was previously analyzed in the Reactor Safety Study (RSS) (Ref. 4.3). The RSS calculated a total point estimate core damage frequency from internal events of 2.6E-5 per year. This study calculated a total median core damage frequency from internal events of 1.9E-6 per year with a corresponding mean value of 4.5E-6. For a detailed discussion of, and insights into, the comparison between this study and the RSS, see Chapter 8.

4.2.1.1 Internally Initiated Accident Sequences

A detailed description of accident sequences important at the Peach Bottom plant is provided in Reference 4.1. For this summary report, the accident sequences described in that report have been grouped into four summary plant damage states. These are:

- Station blackout,
- Anticipated transient without scram (ATWS),
- Loss-of-coolant accidents (LOCAs), and
- Transients other than station blackout and ATWS.

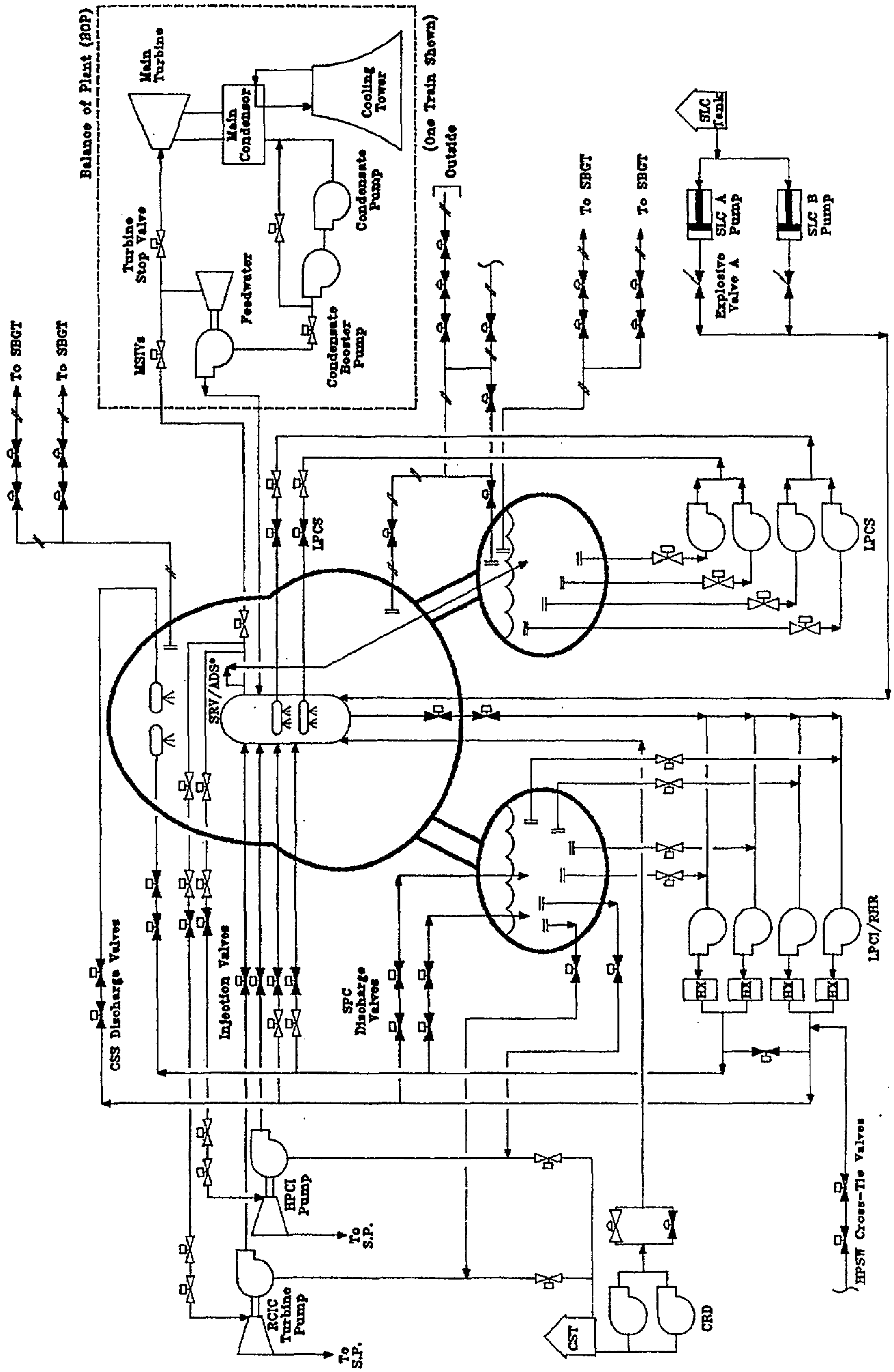
The relative contributions of these groups to mean internal-event core damage frequency at Peach Bottom are shown in Figure 4.3. From Figure 4.3, it may be seen that station blackout sequences as a class are the largest contributor to mean core damage frequency. It should be noted that the plant configuration (as analyzed for this study) does not reflect modifications that may be required in response to the station blackout rule.

Within the general class of station blackout accidents, the more probable combinations of failures leading to core damage are:

- Loss of onsite and offsite ac power results in the loss of all core cooling systems (except high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC), both of which are ac independent in the short term) and all containment heat removal systems. HPCI or RCIC (or both) systems function but ultimately fail at approximately 10 hours because of battery depletion or other late failure modes (e.g., loss of room cooling effects). Core damage results in approximately 13 hours as a result of coolant boiloff.
- Loss of offsite power occurs followed by a subsequent failure of all onsite ac power. The diesel generators fail to start because of failure of all the vital batteries. Without ac and dc power, all core cooling systems (including HPCI and RCIC) and all containment heat removal systems fail. Core damage begins in approximately 1 hour as a result of coolant boiloff.
- Loss of offsite power occurs followed by a subsequent failure of a safety relief valve to reclose. All onsite ac power fails because the diesel generators fail to start and run from a variety of faults. The loss of all ac power fails most of the core cooling systems and all the containment heat removal systems. HPCI and RCIC (which are ac independent) are available and either or both initially function

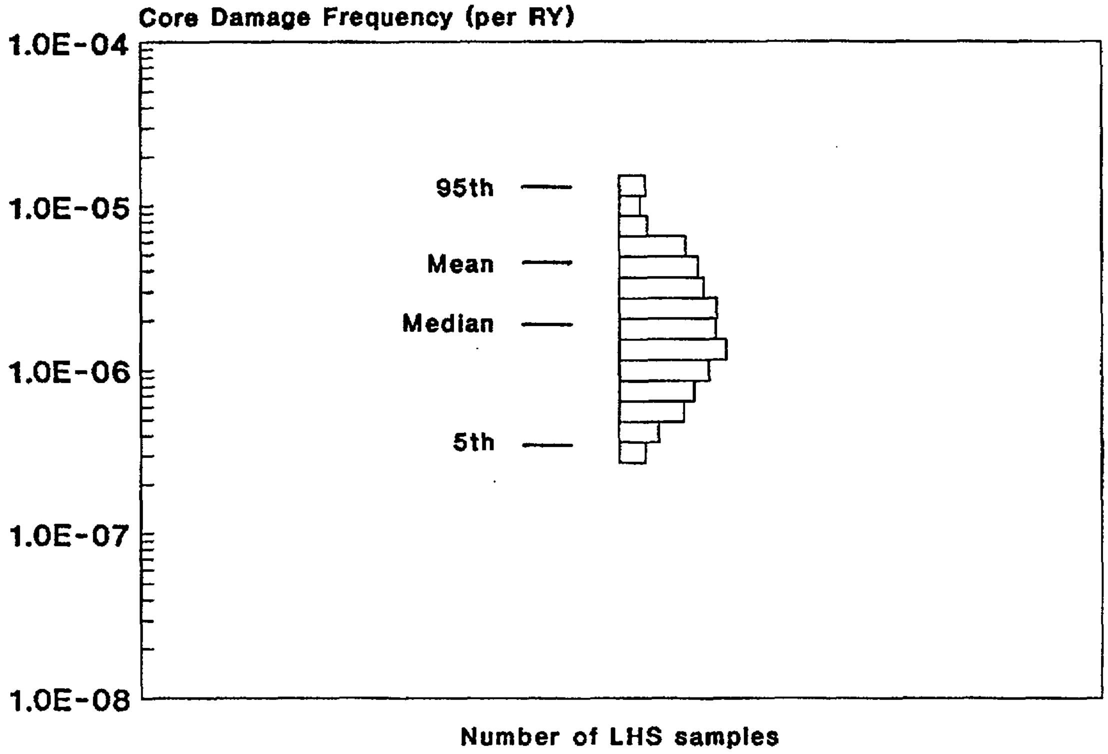
Table 4.1 Summary of design features: Peach Bottom Unit 2.

1. Coolant Injection Systems	a.	High-pressure coolant injection system provides coolant to the reactor vessel during accidents in which system pressure remains high, with 1 train and 1 turbine-driven pump.
	b.	Reactor core isolation cooling system provides coolant to the reactor vessel during accidents in which system pressure remains high, with 1 train and 1 turbine-driven pump.
	c.	Low-pressure core spray system provides coolant to the reactor vessel during accidents in which vessel pressure is low, with 2 trains and 4 motor-driven pumps.
	d.	Low-pressure coolant injection system provides coolant to the reactor vessel during accidents in which vessel pressure is low, with 2 trains and 4 pumps.
	e.	High-pressure service water crosstie system provides coolant makeup source to the reactor vessel during accidents in which normal sources of emergency injection have failed (low RPV pressure), with 1 train and 4 pumps for crosstie.
	f.	Control rod drive system provides backup source of high-pressure injection, with 2 pumps/210 gpm (total)/1,100 psia.
	g.	Automatic depressurization system for depressurizing the reactor vessel to a pressure at which the low-pressure injection systems can inject coolant to the reactor vessel: 5 ADS relief valves/capacity 820,000 lb/hr. In addition, there are 6 non-ADS relief valves.
2. Key Support Systems	a.	dc power with up to approximately 10–12-hour station batteries.
	b.	Emergency ac power from 4 diesel generators shared between 2 units.
	c.	Emergency service water provides cooling water to safety systems and components shared by 2 units.
3. Heat Removal Systems	a.	Residual heat removal/suppression pool cooling system to remove heat from the suppression pool during accidents, with 2 trains and 4 pumps.
	b.	Residual heat removal/shutdown cooling system to remove decay heat during accidents in which reactor vessel integrity is maintained and reactor at low pressure, with 2 trains and 4 pumps.
	c.	Residual heat removal/containment spray system to suppress pressure and remove decay heat in the containment during accidents, with 2 trains and 4 pumps.
4. Reactivity Control Systems	a.	Control rods.
	b.	Standby liquid control system, with 2 parallel positive displacement pumps rated at 43 gpm per pump, but each with 86 gpm equivalent because of the use of enriched boron.
5. Containment Structure	a.	BWR Mark I.
	b.	0.32 million cubic feet.
	c.	56 psig design pressure.
6. Containment Systems	a.	Containment venting—drywell and wetwell vents used when suppression pool cooling and containment sprays have failed to reduce primary containment pressure.



*Typical arrangement (5 ADS SRVs and 6 non-ADS SRVs)

Figure 4.1 Peach Bottom plant schematic.



Note: As discussed in Reference 4.4, 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).

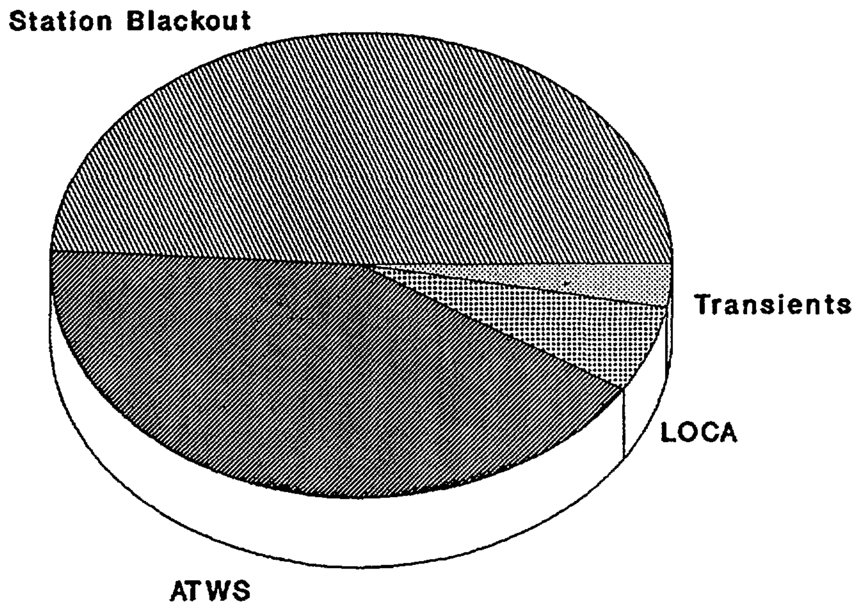
Figure 4.2 Internal core damage frequency results at Peach Bottom.

Table 4.2 Summary of core damage frequency results: Peach Bottom.*

	5%	Median	Mean	95%
Internal Events	3.5E-7	1.9E-6	4.5E-6	1.3E-5
Station Blackout	8.3E-8	6.2E-7	2.2E-6	6.0E-6
ATWS	3.1E-8	4.4E-7	1.9E-6	6.6E-6
LOCA	2.5E-9	4.4E-8	2.6E-7	7.8E-7
Transient	6.1E-10	1.9E-8	1.4E-7	4.7E-7
External Events**				
Seismic (LLNL)	5.3E-8	4.4E-6	7.7E-5	2.7E-4
Seismic (EPRI)	2.3E-8	7.1E-7	3.1E-6	1.3E-5
Fire	1.1E-6	1.2E-5	2.0E-5	6.4E-5

*Note: As discussed in Reference 4.4, 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).

**See "Externally Initiated Accident Sequences" in Section 4.2.1.2 for discussion.



Total Mean Core Damage Frequency: 4.5E-6

Figure 4.3 Contributors to mean core damage frequency from internal events at Peach Bottom.

but ultimately fail at approximately 10 hours because of battery depletion or other late failure modes (e.g., loss of room cooling effects). Core damage results in 10 to 13 hours as a result of coolant boiloff.

Within the general class of anticipated transient without scram accidents, the more probable combinations of failures leading to core damage are:

- Transient (e.g., loss of feedwater) occurs followed by a failure to trip the reactor because of mechanical faults in the reactor protection system (RPS) and closure of the main steam isolation valves (MSIVs). The standby liquid control system (SLCS) does not function (primarily because of operator failure to actuate), but the HPCI does start. However, increased suppression pool temperatures fail the HPCI. Low-pressure coolant injection (LPCI) is unavailable and all core cooling is lost. Core damage occurs in approximately 20 minutes to several hours, depending on the time at which the LPCI fails because of different LPCI failure modes.

- Transient occurs followed by a failure to scram (mechanical faults in the RPS) and closure of the MSIVs. SLCS is initiated but HPCI fails to function because of random faults. The operator fails to depressurize after HPCI failure and therefore the low-pressure core cooling systems cannot inject. Core damage occurs in approximately 15 minutes.

Within the general class of LOCAs, the more probable combination of failures leading to core damage is:

- A medium-size LOCA (i.e., break size of approximately 0.004 to 0.1 ft^2) occurs. HPCI works initially but fails because of low steam pressure. The low-pressure core cooling systems fail to actuate primarily because of miscalibration faults of the pressure sensors, which do not "permit" the injection valves to open. All core cooling is lost and core damage occurs in approximately 1 to 2 hours following the initiating event.

4.2.1.2 Externally Initiated Accident Sequences

A detailed description of accident sequences initiated by external events important at the Peach Bottom plant is provided in Part 3 of Reference 4.1. The accident sequences described in that reference have been grouped into two main types for this study. These are:

- Seismic, and
- Fire.

A scoping study has also been performed to assess the potential effects of other externally initiated accidents (Ref. 4.1, Part 3). This analysis indicated that the following external-event sources could be excluded based on the low frequency of the initiating event:

- Aircraft crashes,
- Hurricanes,
- Tornados,
- Internal flooding, and
- External flooding.

1. Seismic Accident Frequency Analysis

The relative contribution of classes of seismically and fire-initiated accidents to the total mean frequency of externally initiated core damage accidents is provided in Figure 4.4. As may be seen, the dominant seismic scenarios are transient (38%) and LOCA sequences (27%) with the other contributors being substantially less. For these two seismic accident initiators, the more probable combinations of system failures are:

- The transient sequence results from seismically induced failure of ceramic insulators in the switchyard causing loss of offsite power (LOSP) in conjunction with loss of onsite ac power. This latter results primarily from loss of the emergency service water (ESW) system (which provides the jacket cooling for the emergency diesel generators) and/or direct failures of 4 kV buses or the diesel generators themselves. The vast majority of failures are seismically induced.
- The large LOCA sequence is initiated by postulated seismically induced failures of the supports on the recirculation pumps. Core damage results from this initiator in conjunction with seismically induced failures of the low-pressure injection systems. The latter requires ac power, and the dominant sources of failure of onsite ac power are the ESW or emergency diesel generator seismic failures as discussed above.

As discussed in Chapter 2, the seismic analysis in this report made use of two sets of hazard curves from Lawrence Livermore National Laboratory (LLNL) (Ref. 4.5) and the Electric Power Research Institute (EPRI) (Ref. 4.6). The differences between the seismic core damage frequencies shown in Table 4.2 for the LLNL and the EPRI cases are due entirely to the differences between the two sets of hazard curves. That is, the system models, failure rates, and success logic were identical for both estimates.

The seismic hazard associated with the curves developed by EPRI was significantly less than that of the LLNL curves. Differences between these curves result primarily from differences between the methodology and assumptions used to develop the hazard curves. In the LLNL program, considerable emphasis was placed on a wide range of uncertainty in the ground-motion attenuation models, while a relatively coarse set of seismic tectonic provinces was used in characterizing each site. By contrast, in the EPRI program considerable emphasis was placed on a fine zonation for the tectonic provinces, and very little uncertainty in the ground-motion attenuation was considered. In any case, it is the difference between the two sets of hazard curves that causes the differences between the numeric estimates in Table 4.2.

2. Fire Accident Frequency Analysis

The fire-initiated accident frequency analyses performed for this report considered the impact of fires beginning in a variety of separate locations within the plant. Those locations found to be most important were:

- Emergency switchgear rooms,
- Control room, and
- Cable-spreading room.

No other plant locations contributed more than 1.0E–8 per year to the core damage frequency.

Fires in the cable-spreading room are assumed to require manual plant trip and to fail the high-pressure injection and depressurization systems, namely: high pressure core injection (HPCI), reactor core isolation cooling (RCIC), control rod drive (CRD), and automatic depressurization systems (ADS). In each case, the failure occurs because of fire damage to the control cables.

Fires in the emergency switchgear rooms failed offsite power and in some instances portions of the emergency service water system, and core damage occurs because of a station blackout sequence involving additional random failures of the emergency service water system (which provides jacket cooling to the diesel generators).

Finally, two fire scenarios were identified for the control room, both of which involve manual plant trip and abandonment of the control room. One scenario involved random failure of the RCIC system and a reasonable probability that the operators fail to recover the plant using HPCI or ADS in conjunction with LPCI from the remote shutdown panel. The other scenario failed the RCIC system because of a fire in its control cabinet but allowed for recovery from the remote shutdown panel.

4.2.2 Important Plant Characteristics (Core Damage Frequency)

Characteristics of the Peach Bottom plant design and operation that have been found to be important in the analysis of core damage frequency include:

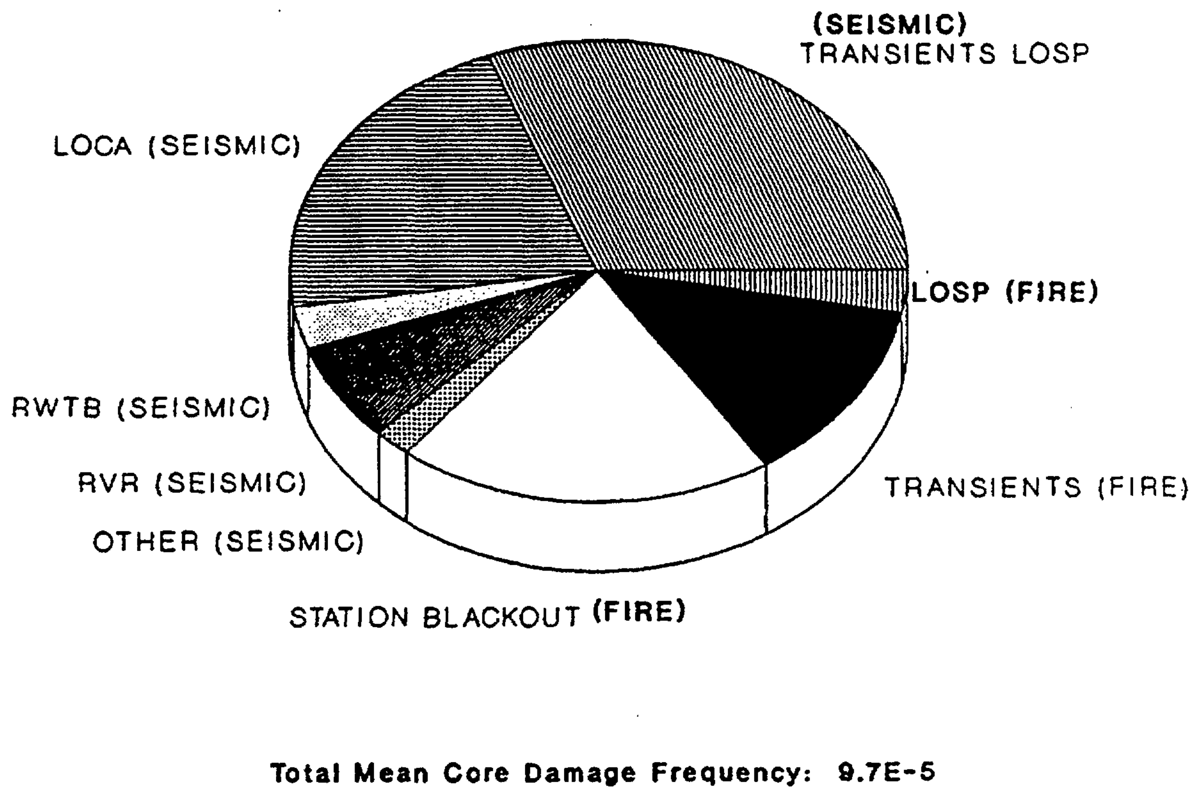


Figure 4.4 Contributors to mean core damage frequency from external events (LLNL hazard curve) at Peach Bottom.

1. **High-Pressure Service Water System Crosstie**

   The high-pressure service water (HPSW) system, if the reactor vessel has been depressurized, can inject raw water to the reactor vessel via the residual heat removal injection lines. Most components of HPSW are located outside the reactor building and thus are not affected by any potential severe reactor building environment that could cause other injection systems to fail in some accidents. Therefore, this system offers diversity, as well as redundancy, and affects many different types of sequences. The Peach Bottom operators are trained to use this system and can do so from the control room. An extensive cleanup program would, however, be required after the system is initiated.

2. **Redundancy and Diversity of Water Supply Systems**

   At Peach Bottom, there are many redundant and diverse systems to provide water to the reactor vessel. They include:

High-pressure core injection (HPCI) with 1 pump;

Reactor core isolation cooling (RCIC) with 1 pump;

Control rod drive (CRD) with 2 pumps (both pumps required);

Low-pressure core spray (LPCS) with 4 pumps;

Low-pressure core injection (LPCI) with 4 pumps;

Condensate with 3 pumps; and

High-pressure service water (HPSW) with 4 pumps.

Because of this redundancy of systems, LOCAs and transients other than station blackout and ATWS are small contributors to the core damage frequency.

CRD, condensate, and HPSW pumps are located outside the reactor building (generally away from potentially severe environments) and represent excellent secondary high- and low-pressure coolant systems if normal injection systems fail. These systems are not available during station blackout.

3. **Redundancy and Diversity of Heat Removal Systems**

At Peach Bottom, there are several diverse means for heat removal. These systems are:

Main steam/feedwater system;

Suppression pool cooling mode of residual heat removal (RHR);

Shutdown cooling mode of RHR;

Containment spray system mode of RHR; and

Containment venting.

This diversity has greatly reduced the importance of transients with long-term loss of heat removal.

4. **Diesel Generators**

Peach Bottom is a two-unit site with four emergency diesels shared between the two units. One diesel can supply the necessary power for both units. DC power to start the diesels is supplied from vital dc station batteries. The four emergency diesels share a common service water system that provides oil cooling, jacket, and air cooling. The Peach Bottom emergency diesels historically have had a failure-to-start probability that is much better than the industry average, e.g., a factor of ~10 lower failure probability.

5. **Battery Capacity**

Philadelphia Electric Company (PECo) has performed analyses of the battery life based on the current station blackout procedures. PECo estimates that the station batteries at Peach Bottom are capable of lasting at least 12 hours in a station blackout. They have revised their station blackout procedure to include load shedding in order to ensure a longer period of injection and accident monitoring. The ability to ensure availability for 12 hours reduces the frequency of core damage resulting from station blackout accident sequences.

6. **Emergency Service Water (ESW) System**

The ESW system provides cooling water to selected equipment during a loss of offsite power. The system has two full capacity self-cooled pumps whose suction is from the Conowingo pond and a backup third pump with a separate water source. Failure of the ESW system would quickly fail operating diesel generators and potentially fail the low-pressure core spray (LPCS) pumps and the RHR pumps. The HPCI pumps and RCIC pumps would fail (in the long term) from a loss of their room cooling after a loss of the ESW system.

It should be noted that there is an outstanding issue regarding the need for ESW that involves whether or not the LPCS/RHR pumps actually require ESW cooling. PECo has stated that these pumps are designed to operate with working fluid temperatures approaching 160°F without pump cooling. This implies that in scenarios where the ESW system has been lost, these pumps could still operate; some RHR pumps would be placed in the suppression pool cooling mode and therefore keep the working fluid at less than 160°F. It is felt that there is significant validity to these arguments. However, because it is uncertain whether the suppression pool water can be maintained below 160°F in some sequences and whether PECo has properly accounted for pump heat addition to the system, the analysis summarized here assumes these LPCS/RHR pumps will fail upon loss of ESW cooling.

7. **Automatic and Manual Depressurization System**

The automatic depressurization system (ADS) is designed to depressurize the reactor vessel to a pressure at which the low-pressure injection systems can inject coolant. The ADS consists of five safety relief valves capable of being manually opened. The operator may manually initiate the ADS or may depressurize the reactor vessel, using the six additional relief valves that are not connected to the ADS logic. The ADS valves are located inside the containment; however, the instrument nitrogen and the dc power required to operate the valves are supplied from outside the containment.

8. **Standby Liquid Control (SLC) System**

The SLC system provides a backup method that is redundant but independent of the control rods to establish and maintain the reactor subcritical. The suction for the SLC system comes from a control tank that has sodium pentaborate in solution with demineralized water. Most of the SLC system is located in the reactor building outside the drywell. Local access to the SLC system could be affected by containment failure or containment venting.

9. **Venting Capability**

The primary containment venting system at Peach Bottom is used to prevent containment pressure limits from being exceeded. There are several vent paths:

- 2-inch torus vent to standby gas treatment (SBGT),
- 6-inch integrated leak rate test (ILRT) pipe from the torus,
- 18-inch torus vent path,
- 18-inch torus supply path,
- 2-inch drywell vent to SBGT,
- Two 3-inch drywell sump drain lines,
- 6-inch ILRT line from drywell,
- 18-inch drywell vent path, and
- 18-inch drywell supply path.

The types of sequences on which venting has the most effect are transients with long-term loss of decay heat removal. The chance of survival of the containment is increased with venting; therefore, the core damage frequency from such sequences is reduced.

If the reactor is at decay heat loads, venting using the 6-inch ILRT line or equivalent as a minimum is sufficient to lessen the containment pressure. However, in an ATWS sequence, three to four of the large 18-inch vent pathways need to be used in order to achieve the same effect. It is preferable to use a vent pathway from the torus rather than from the drywell because of the scrubbing of radioactive material coming through the suppression pool.

It is significant to note that the 6-inch ILRT line is a solid pipe rather than ductwork, so that venting by means of this pipe does not create a severe environment within the reactor building; use of the 18-inch lines will result in failure of the ductwork and severe environments within the reactor building.

10. **Location of Control Rod Drive (CRD) Pumps**

The CRD pumps at Peach Bottom are not located in the reactor building (like most plants) but are in the turbine building. Therefore, in a severe accident where severe environments are sometimes created, the CRD pumps are not subjected to these environments and can continue to operate.

4.2.3 Important Operator Actions

The emergency operating procedures (EOPs) at Peach Bottom direct the operator to perform certain actions depending on the plant conditions or symptoms (e.g., reactor vessel level below top of active fuel). Different accident sequences can have similar symptoms and therefore the same "recovery" actions. The operator actions that either are important in reducing accident frequencies or are contributing to accident frequencies are discussed and can apply to many different accident sequences.

The quantification of these human failure events was based on an abbreviated version of the THERP method (Ref. 4.7). These failure events include the following:

- Actuate core cooling

  In an accident where feedwater is lost (which includes condensate), the reactor vessel water level starts to decrease. When Level 2 is reached, HPCI and RCIC should be automatically actuated. If Level 1 is reached, the automatic depressurization system (ADS) should be actuated with automatic actuation of the low-pressure core spray (LPCS) and low-pressure coolant injection (LPCI). If these systems fail to actuate, the operator can attempt to manually actuate them from the control room. In addition, the operator can attempt to recover the power conversion system (PCS) (i.e., feedwater) or manually initiate control rod drive (CRD) (i.e., put CRD in its enhanced flow mode). If automatic depressurization failure was one of the faults, the operator can manually depressurize so that LPCS and LPCI can inject. Lastly, the operator also has the option to align the HPSW to LPCI for another core cooling system.

- Establish containment heat removal

  Besides core cooling, the operator must also establish containment heat removal (CHR). Without CHR, the potential exists for operating core cooling systems to fail. If an accident occurs, the EOPs direct the operator to initiate the suppression pool cooling mode of residual heat removal (RHR) after the suppression pool temperature reaches 95°F. The operator closes the LPCI injection valves and the heat exchanger bypass valves and opens the suppression pool discharge valves. He also ensures that the proper service water system train is operating. With suppression pool cooling (SPC) functioning, CHR is being performed. If system faults preclude the use of SPC, the operator has other means to provide CHR. He can actuate other modes of RHR such as shutdown cooling or containment spray; or the operator can vent the containment to remove the heat.

- Restore service water

  Many of the components/systems require cooling water from the emergency service water (ESW) system in order to function. If the ESW pumps fail, the operator can manually start the emergency cooling water pump, which is a backup to the ESW pumps.

Specifically for station blackout, there are certain actions that can be performed by the operating crew:

- Recovering ac power

  Station blackout is caused by the loss of all ac power, i.e., both offsite and onsite power. Restoring offsite power or repairing the diesel generators was included in the analysis. The quantification of these human failure events was derived from historical data (i.e., actual time required to perform these repairs) and not by performing a human reliability analysis on these events.

Transients where reactor trip does not occur (i.e., ATWS) involve accident sequences where the phenomena are more complex. The operator actions were evaluated in more detail (using the SLIM-MAUD* method performed by Brookhaven National Laboratory (Ref. 4.8)) than for the regular transients. These actions include the following:

- Manual scram

  A transient that demands the reactor to be tripped occurs, but the reactor protection system (RPS) fails from electrical faults. The operator can then manually trip the reactor by first rotating the collar on the proper scram buttons and then depressing the buttons, or he can put the reactor mode switch in the "shutdown" position.

- Insert rods manually

  If the electrical faults fail both the RPS and the manual trip, the operator can manually insert the control rods one at a time.

- Actuate standby liquid control (SLC)

  With the reactor not tripped, reactor power remains high; the reactor core is not at decay heat levels. This can present problems since the CHR systems are only designed to decay heat removal capacity. However, the SLC system (manually activated) injects sodium pentaborate that reduces reactor power to decay heat levels. The EOPs direct the operator to actuate SLC if the reactor power is above 3 percent and before the suppression pool temperature reaches 110°F. The operator obtains the SLC keys (one per pump) and inserts the keys into the switches and turns only one to the "on" position.

- Inhibit automatic depressurization system (ADS)

  In an ATWS condition, the operator is directed to inhibit the ADS if he has actuated SLC. The operator must put both ADS switches in the inhibit mode.

*SLIM-MAUD is a computer algorithm for transforming man-man and man-machine information into probability statements.

- Manually depressurize reactor

  If the high-pressure coolant injection (HPCI) fails, inadequate high-pressure core cooling occurs. Because the ADS was inhibited, when Level 1 is reached, ADS will not occur and the operator must manually depressurize so that low-pressure core cooling can inject.

4.2.4 Important Individual Events and Uncertainties (Core Damage Frequency)

As discussed in Chapter 2, the process of developing a probabilistic model of a nuclear power plant involves the combination of many individual events (initiators, hardware failures, operator errors, etc.) into accident sequences and eventually into an estimate of the total frequency of core damage. After development, such a model can also be used to assess the relative importance and contribution of the individual events. The detailed studies underlying this report have been analyzed using several event importance measures. The results of the analyses using two measures, "risk reduction" and "uncertainty" importance, are summarized below.

- Risk (core damage frequency) reduction importance measure (internal events)

  The risk-reduction importance measure is used to assess the change in core damage frequency as a result of setting the probability of an individual event to zero. Using this measure, the following individual events were found to cause the greatest reduction in core damage frequency if their probabilities were set to zero:

  - Mechanical failure of the reactor protection system. The core damage frequency would be reduced by approximately 52 percent.
  - Transient initiators with the power conversion system available. The core damage frequency would be reduced by approximately 47 percent.
  - Loss of offsite power initiating event. The core damage frequency would be reduced by approximately 39 percent.
  - Operator failure to restore the standby liquid control system after testing. The core damage frequency would be reduced by approximately 25 percent.
  - Operator failure to initiate emergency heat sink. The core damage frequency would be reduced by approximately 17 percent.
  - Operator failure to actuate standby liquid control system. The core damage frequency would be reduced by approximately 16 percent.
  - Operator miscalibrates reactor pressure sensors. The core damage frequency would be reduced by approximately 12 percent.

  Note that the top risk-reduction events do not necessarily appear in the most frequent sequences since the latter sequences may result from the cumulative influence of many lesser contributors.

- Uncertainty importance measure (internal events)

  A second importance measure used to evaluate the core damage frequency analysis results is the uncertainty importance measure. For this measure, the relative contribution of the uncertainty of individual events to the uncertainty in total core damage frequency is calculated. Using this measure, the following events were found to be most important:

  - Mechanical failure of the reactor protection system.
  - Failure of the diesel generators to continue to run once started.
  - Loss of offsite power or transients with the power conversion system available.
  - Miscalibration of the reactor pressure sensors by the operator.
  - Operator failure to restore the standby liquid control system after testing.

4.3 Containment Performance Analysis

4.3.1 Results of Containment Performance Analysis

The Peach Bottom Mark I containment design concept consists of a pressure-suppression containment system that houses the reactor vessel, the reactor coolant recirculating loops, and other branch connections to the reactor coolant system. The containment design consists of a light-bulb-shaped drywell and a water-filled toroidal-shaped suppression pool. Both the drywell and the suppression pool are freestanding steel shells with the drywell region backed by a reinforced concrete structure. The containment system has a volume of 320,000 cubic feet and is designed to withstand a peak pressure of 56 psig resulting from a primary system loss-of-coolant accident. The estimated mean failure pressure for Peach Bottom's containment system is 148 psig, which is very similar to that for large PWR containment designs. However, its small free volume relative to other containment types significantly limits its capacity to accommodate noncondensible gases generated in severe accident scenarios in addition to increasing its potential to come into contact with molten core material. The complexity of the events occurring in severe accidents has made predictions of when and where Peach Bottom's containment would fail heavily reliant on the use of expert judgment to interpret and supplement the limited data available.

The potential for early containment failure (before or within roughly 2 hours after reactor vessel breach) is of principal concern in Peach Bottom's risk analysis. For the Peach Bottom Mark I type of containment, the principal mechanisms that can cause its early failure are (1) drywell shell meltthrough due to its interaction with the molten core material released from the breached reactor pressure vessel, (2) overpressure failure of the drywell due to rapid direct containment heating following reactor vessel breach, and (3) stretching of the drywell head bolts (due to internal pressurization) causing a direct leakage path from the system. Possible overpressure failures due to hydrogen combustion effects are of negligible probability for Peach Bottom since the containment is inerted. In addition to the early modes of containment failure, core damage sequences can also result in late containment failure or no containment failure at all.

The results of the Peach Bottom containment analysis are summarized in Figures 4.5 and 4.6. Figure 4.5 contains a display of information in which the conditional probabilities of 10 containment-related accident progression bins; e.g., V.B-early WWF - >200, are presented for each of six plant damage states, such as station blackout. This information indicates that, on a plant damage state frequency-weighted average,* the mean conditional probability from internally initiated accidents of: (1) early wetwell failure is about 0.03, (2) early drywell failure is about 0.52, (3) late failure of either the wetwell or drywell is about 0.04, and (4) no containment failure is about 0.27. Figure 4.6 further displays the conditional probability distribution of early containment failure for each plant damage state, thereby providing the estimated range of uncertainties in these containment failure predictions. The important conclusions that can be drawn from the information in these two figures are: (1) there is a high mean probability (i.e., 50%) that the Peach Bottom containment will fail early for the dominant plant damage states; (2) early containment failures will primarily occur in the drywell structure resulting in a bypass of the suppression pool's scrubbing effects for radioactive material released after vessel breach; and (3) the principal cause of early drywell failure is drywell shell meltthrough. The data further indicate that the early containment failure probability distributions for most plant damage states are quite broad. Also presented in these displays of containment failure information is evidence that there is a high probability of early containment failure during external events such as fire and earthquakes. Specifically, the seismic analysis indicates that the conditional probability of early containment failure from all causes, i.e., direct containment structural failure or related failure from the effects of a core damage event, could be as high as 0.9.

*Each value in the column in Figure 4.5 labeled "All" is obtained by summing the products of individual accident progression bin conditional probabilities for each plant damage state and the ratio of the frequency of that plant damage state to the total core damage frequency.

Additional discussion on containment performance (for all studied plants) is provided in Chapter 9.

4.3.2 Important Plant Characteristics (Containment Performance)

Characteristics of the Peach Bottom containment design and operation that are important during core damage accidents include:

1. **Containment Inerting**

   The Peach Bottom containment is maintained in an inerted state, i.e., nitrogen filled. This inerted containment condition significantly reduces the chance of hydrogen combustion in the containment, thereby removing a major threat to its failure. However, hydrogen combustion in the reactor building is a possibility for some severe accident sequences.

2. **Drywell Sprays**

   The Peach Bottom drywell contains a spray header that can be used to mitigate the effects of the actions of molten core material on the floor of the drywell. In particular, the spray system may provide sufficient water to prevent the molten core material from coming into contact with the drywell shell and potentially causing its failure.

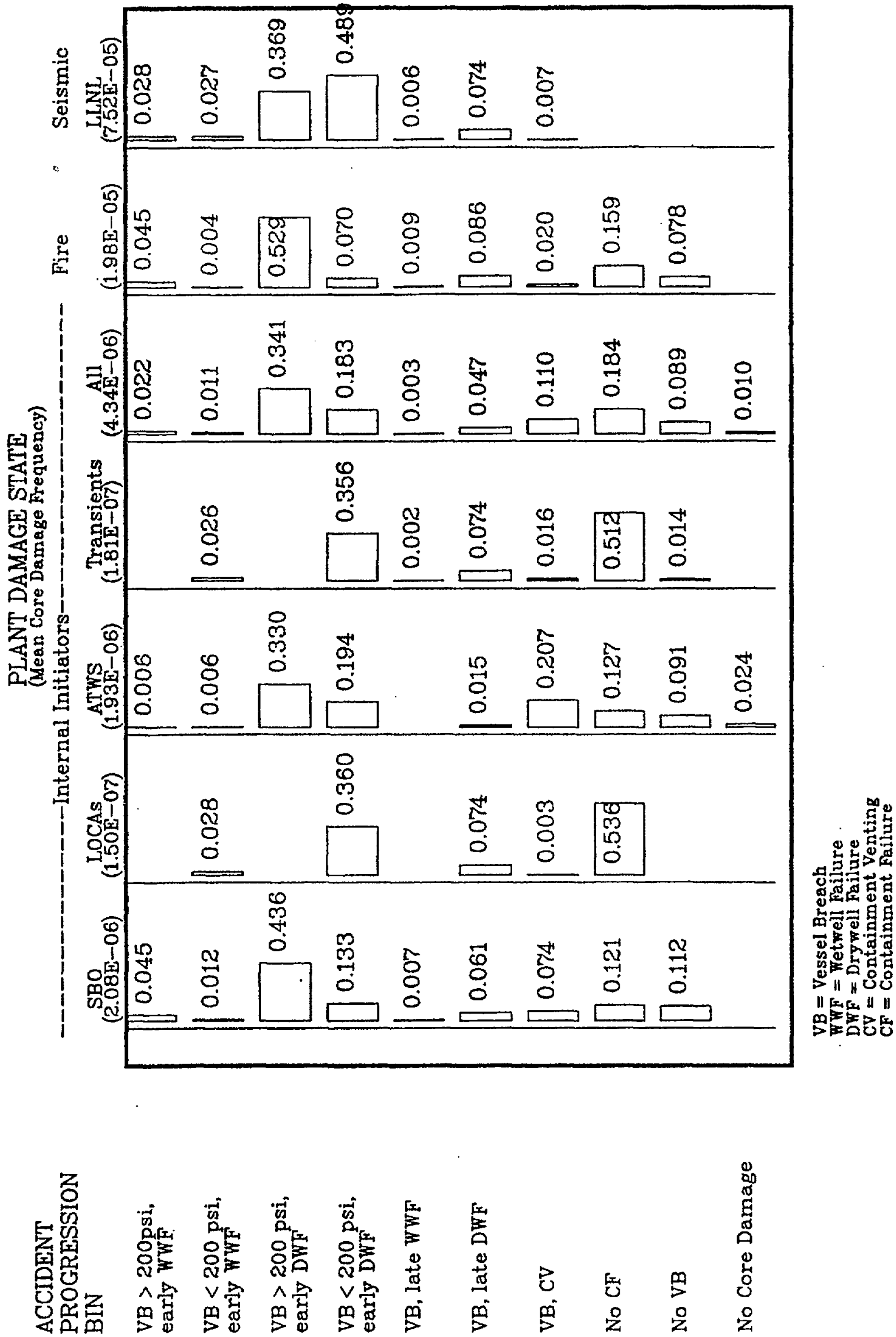


Figure 4.5 Conditional probability of accident progression bins at Peach Bottom.

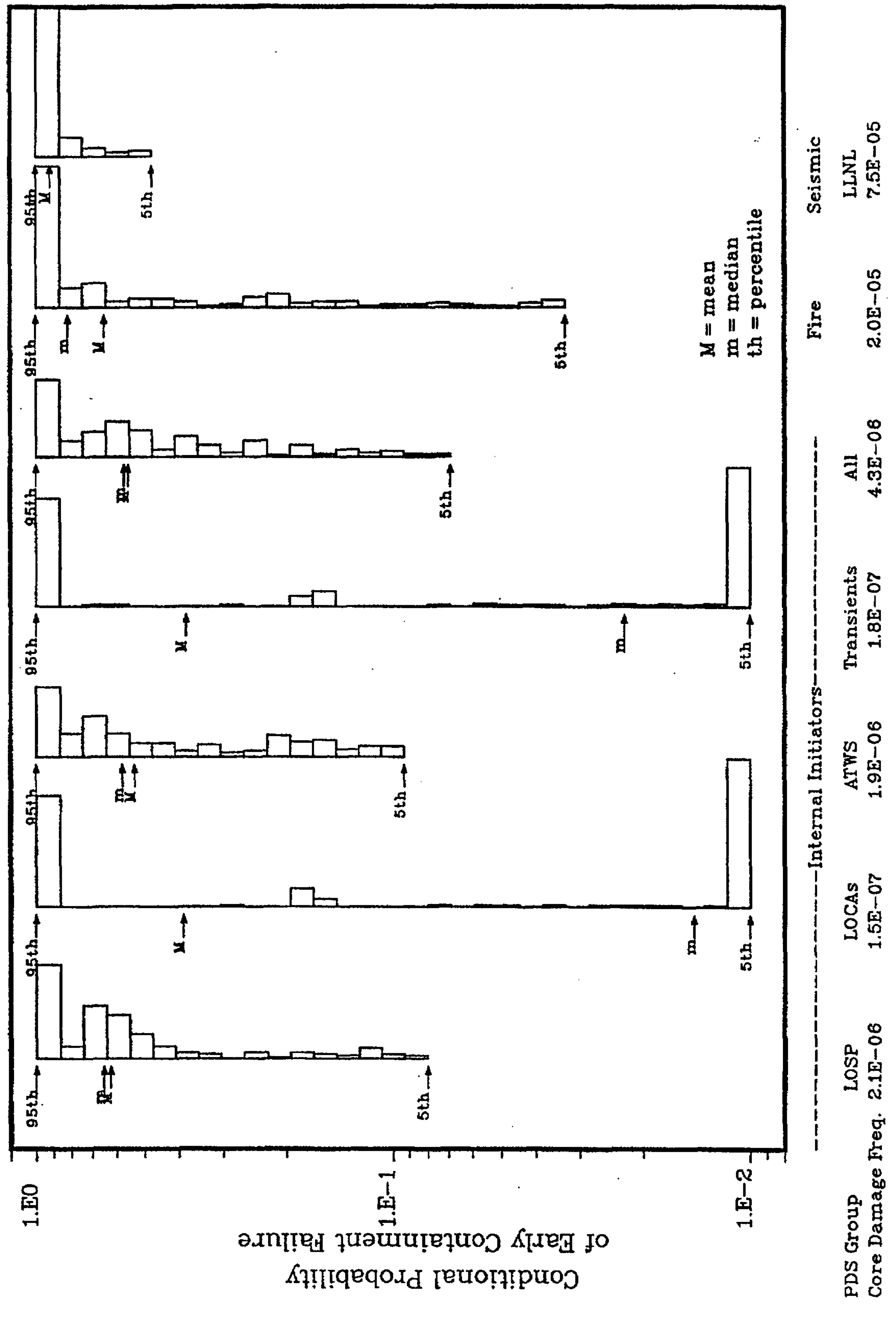


Figure 4.6 Conditional probability distributions for early containment failure at Peach Bottom.

4.4 Source Term Analysis

4.4.1 Results of Source Term Analysis

Failure of the drywell shell following vessel meltthrough is a characteristic of the risk-dominant accident progression bins for the Peach Bottom plant. Figure 4.7 illustrates the source terms for the early failure accident progression bin in which the reactor coolant system is pressurized (> 200 psi) at the time of vessel failure. In comparison with the bypass release that was illustrated for Surry in Figure 3.7, the core fractions of the volatile groups (iodine, cesium, and tellurium) released to the environment are slightly reduced. For the majority of accident sequences in Peach Bottom, the radionuclides released from fuel in-vessel must pass through the suppression pool where substantial decontamination is possible. In sequences where the drywell spray system is operable, the ex-vessel release will also be mitigated by the spray or an overlaying pool of water. Both the in-vessel and ex-vessel releases will receive further attenuation in the reactor building before release to the environment. Even if the decontamination factor of some of these stages is small, the overall effect is to make the likelihood of a very large release quite small.

The Peach Bottom plant has instituted emergency operating procedures to vent the containment in the wetwell region to avoid failure by overpressurization. Figure 4.8 shows the source terms for the accident progression bin in which the containment is vented and no subsequent failure of the containment occurs. The source terms for the volatile radionuclide groups are less than those for the early drywell failure bin discussed previously. In both cases, scrubbing of the in-vessel release by the suppression pool has the principal mitigating influence on the environmental release. The release fractions for the less volatile groups are smaller for the vented accident progression bin but only by approximately a factor of one-half. There are two reasons why the differences between the environmental release of the ex-vessel species for the vented and drywell failure cases are not greater. The decontamination capability of the suppression pool for ex-vessel release, in which the flow is through the downcomers, is somewhat less than for the in-vessel release, which passes through spargers on the safety relief lines. Thus, even though the ex-vessel release must pass through the pool for the vented case, the decontamination factor may be small. The ex-vessel release for the drywell failure accident progression bin will at least be subjected to decontamination in the reactor building and possibly to sprays and scrubbing by an overlaying water layer.

The range of uncertainty in the release for the barium and strontium radionuclide groups is particularly evident. The spread between the mean and median is two orders of magnitude. Although the release is likely to be quite small, the mean value of the release is as high as the mean value for the tellurium release.

Additional discussion on source term perspectives is provided in Chapter 10.

4.4.2 Important Plant Characteristics (Source Term)

1. **Reactor Building**

   The Peach Bottom containment is located within a reactor building. A release of radioactive material to the reactor building will undergo some degree of decontamination before release to the environment. An important consideration in determining the magnitude of building decontamination is whether hydrogen combustion occurs in the building and whether combustion is sufficiently energetic to fail the building. The range of decontamination factors for the reactor building used in the study is from 1.1 to 10 with a median value of 3 for typical accident conditions.

2. **Pressure-Suppression Pool**

   The pressure-suppression pool is particularly effective in the reduction of the in-vessel release component of the source terms for Peach Bottom. The range of decontamination factors used is from 1.2 to 4000 with a median of 80 for flow through the safety relief valve lines.

   The submergence is less and bubble size is larger for flow through the downcomers than for the spargers through which the in-vessel release is most likely to enter the pool. As a result, the decontamination factor for the ex-vessel release or any in-vessel release that passes through the drywell is smaller, ranging from approximately 1 to 90 with a median of 10. Furthermore, the likelihood of failure of the drywell at the time of vessel meltthrough is predicted to be high. For scenarios involving early drywell failure, the suppression pool would be bypassed during the period of core-concrete interaction and radionuclide release.

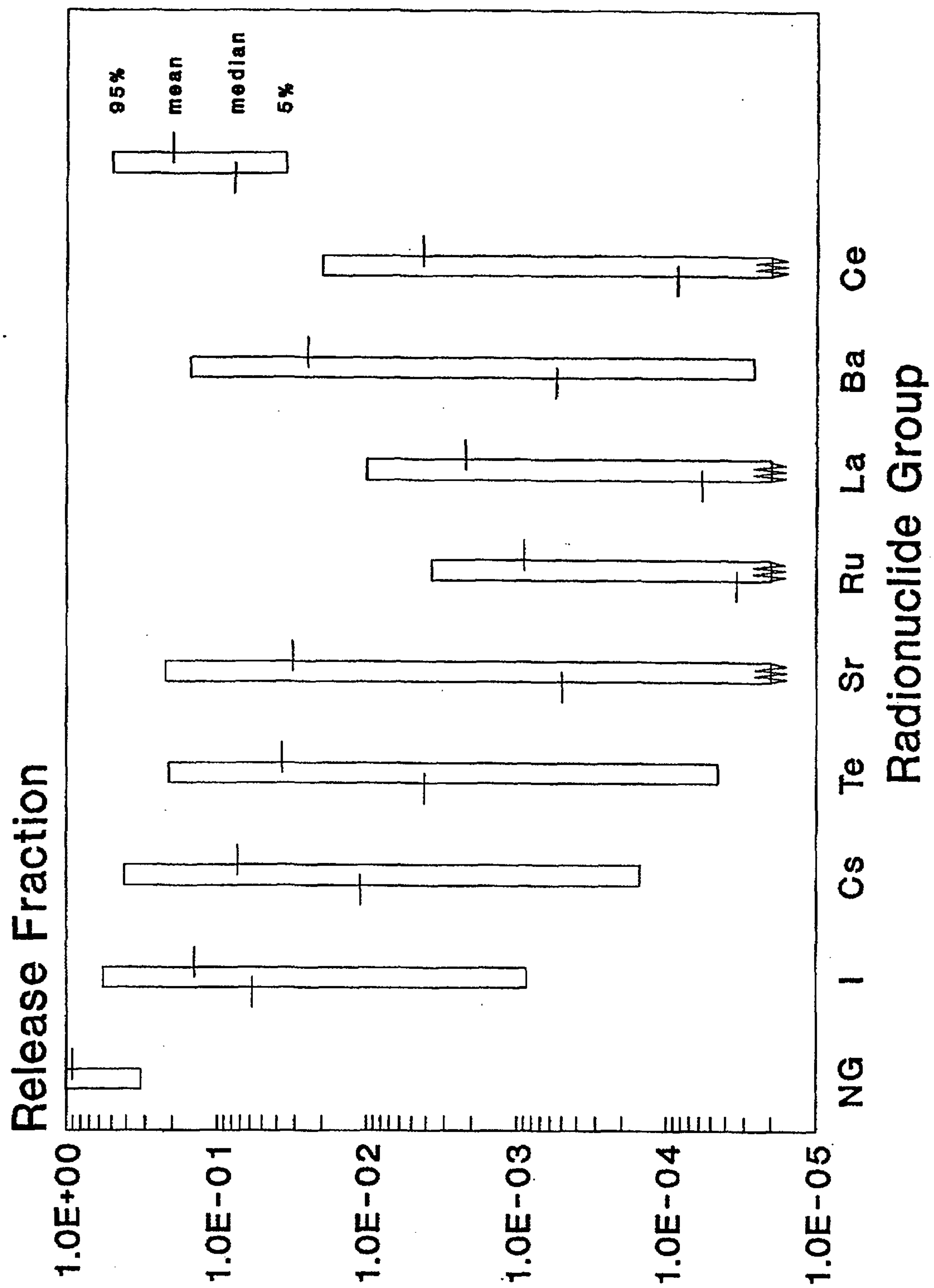


Figure 4.7 Source term distributions for early failure in drywell at Peach Bottom.

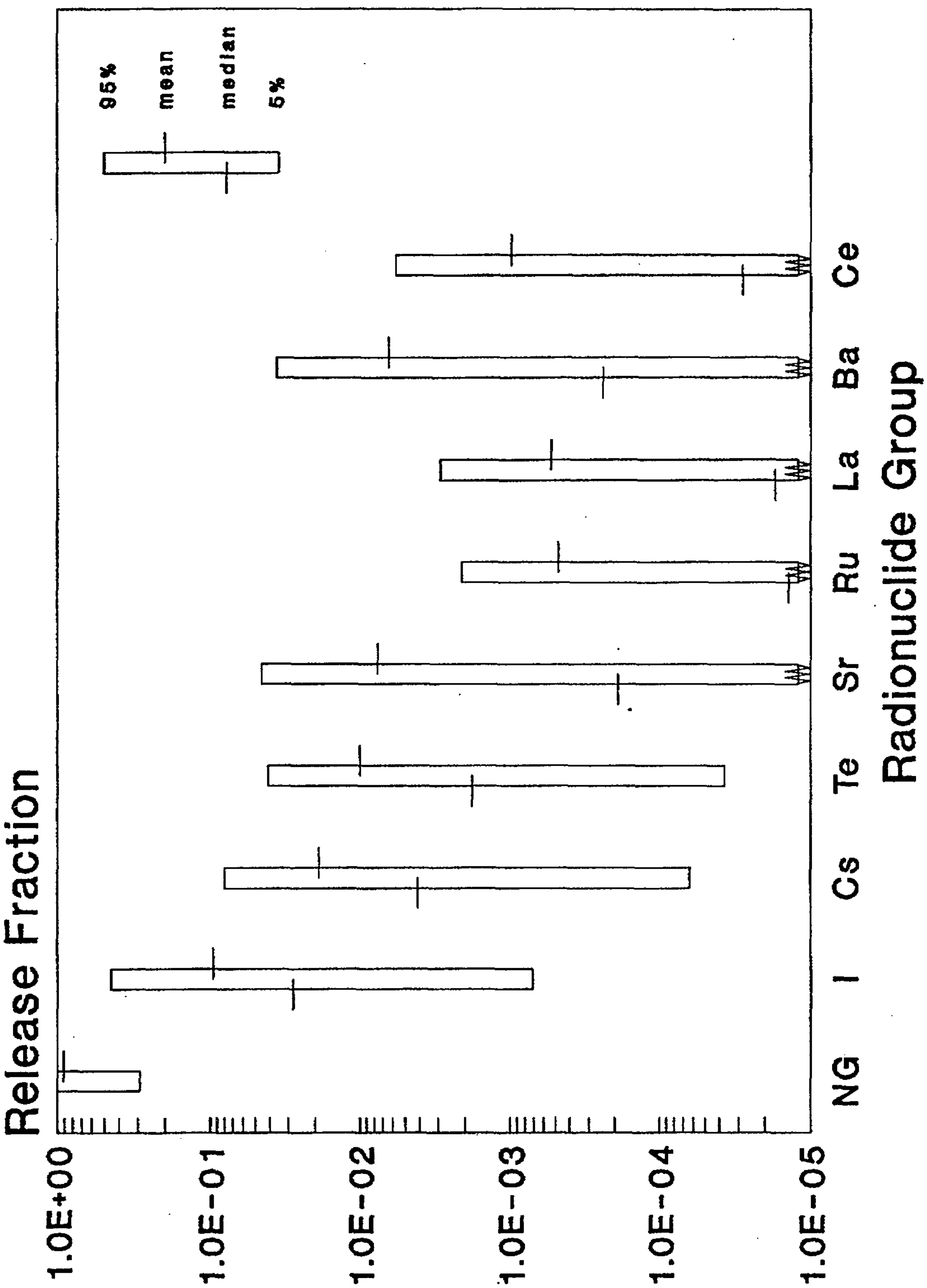


Figure 4.8 Source term distributions for vented containment at Peach Bottom.

3. **Venting**

   The Peach Bottom containment can be vented from the wetwell air space. By preventing containment failure, venting can potentially prevent some scenarios from becoming core damage accidents. In scenarios that proceed to fuel melting, venting can lead to the mitigation of the release of radioactive material to the environment by ensuring that the release passes through the suppression pool. The effect of venting on core damage frequency is described in Chapter 8. Figure 4.8 illustrates the source term characteristics for the venting accident progression bins. Although the source terms are somewhat less than for the early drywell failure accident progression bin, the uncertainties in the release fractions are quite broad. At the high end of the uncertainty range, it is possible that 40 percent of the core inventory of iodine could be released to the environment.

   The effectiveness of venting to mitigate severe accident release of radioactive material is limited in the Peach Bottom analyses because of the high likelihood of early drywell failure, particularly as the result of direct attack of the shell by molten core debris. If direct attack of the containment shell is determined not to lead to failure or if effective means are found to preclude failure, the effectiveness of venting could be greater. However, considering the range of uncertainties in the source term analyses, the predicted consequences of vented accident progression bins are not necessarily minor.

4.5 Offsite Consequence Results

Figures 4.9 and 4.10 display the frequency distributions in the form of graphical plots of the complementary cumulative distribution functions (CCDFs) of four offsite consequence measures—early fatalities, latent cancer fatalities, and the 50-mile and entire site region population exposures (in person-rems). The CCDFs in Figures 4.9 and 4.10 include contributions from all source terms associated with reactor accidents caused by the internal initiating events and fire, respectively. Four CCDFs, namely, the 5th percentile, 50th percentile (median), 95th percentile, and the mean CCDFs, are shown for each consequence measure.

Peach Bottom plant-specific and site-specific parameters were used in the consequence analysis for these CCDFs. The plant-specific parameters included source terms and their frequencies, the licensed thermal power (3293 MWt) of the reactor, and the approximate physical dimensions of the power plant building complex. The site-specific parameters included exclusion area radius (820 meters), meteorological data for 1 full year collected at the site meteorological tower, the site region population distribution based on the 1980 census data, topography (fraction of the area that is land—the remaining fraction is assumed to be water), land use, agricultural practice and productivity, and other economic data for up to 1,000 miles from the Peach Bottom plant.

The consequence estimates displayed in these figures have incorporated the benefits of the following protective measures: (1) evacuation of 99.5 percent of the population within the 10-mile plume exposure pathway emergency planning zone (EPZ), (2) early relocation of the remaining population only from the heavily contaminated areas both within and outside the 10-mile EPZ, and (3) decontamination, temporary interdiction, or condemnation of land, property, and foods contaminated above acceptable levels.

The population density within the Peach Bottom 10-mile EPZ is about 90 persons per square mile. The average delay time before evacuation (after a warning prior to radionuclide release) from the 10-mile EPZ and average effective evacuation speed used in the analyses were derived from information contained in a utility-sponsored Peach Bottom evacuation time estimate study (Ref. 4.9) and the NRC requirements for emergency planning.

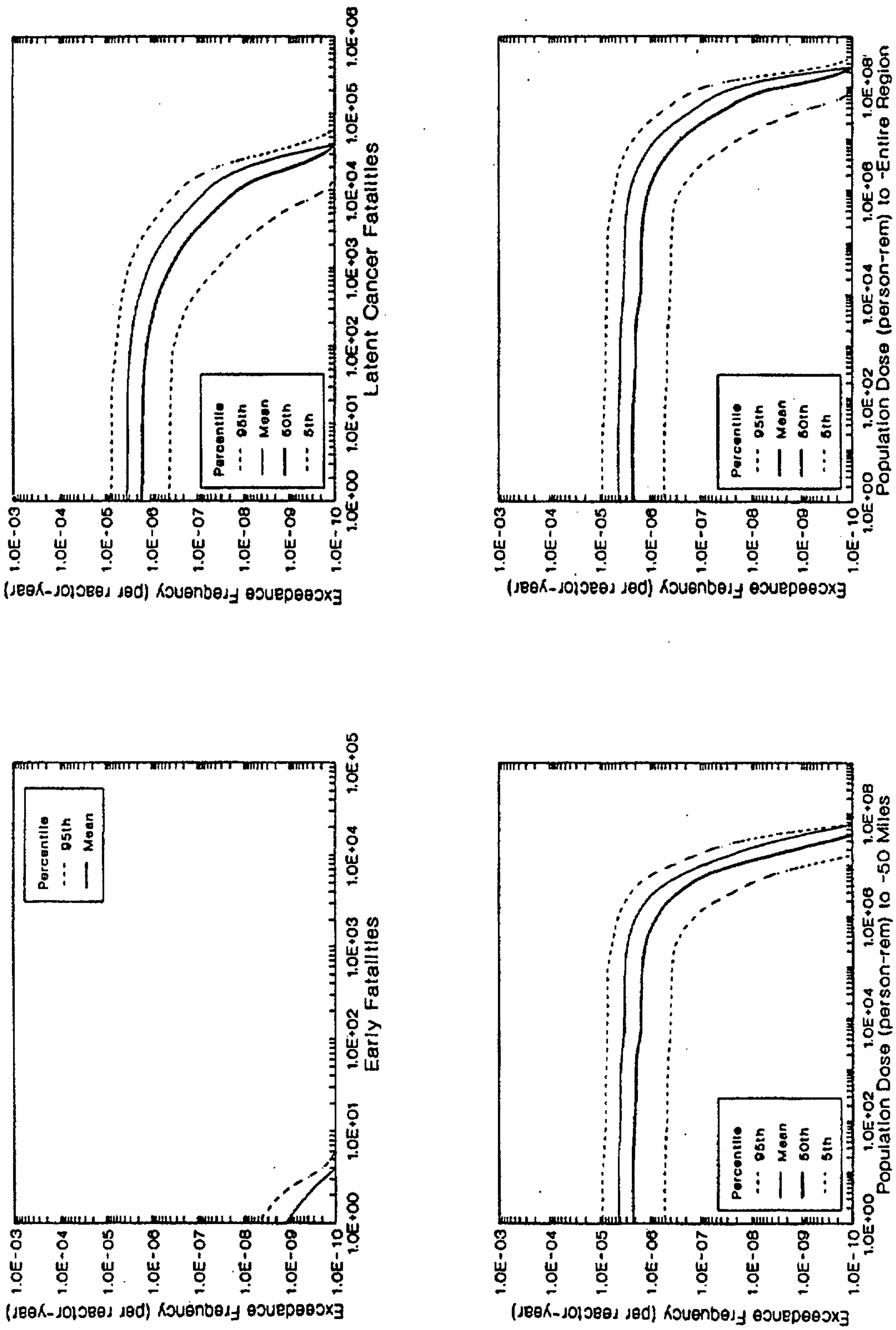
The results displayed in Figures 4.9 and 4.10 are discussed in Chapter 11.

4.6 Public Risk Estimates

4.6.1 Results of Public Risk Estimates

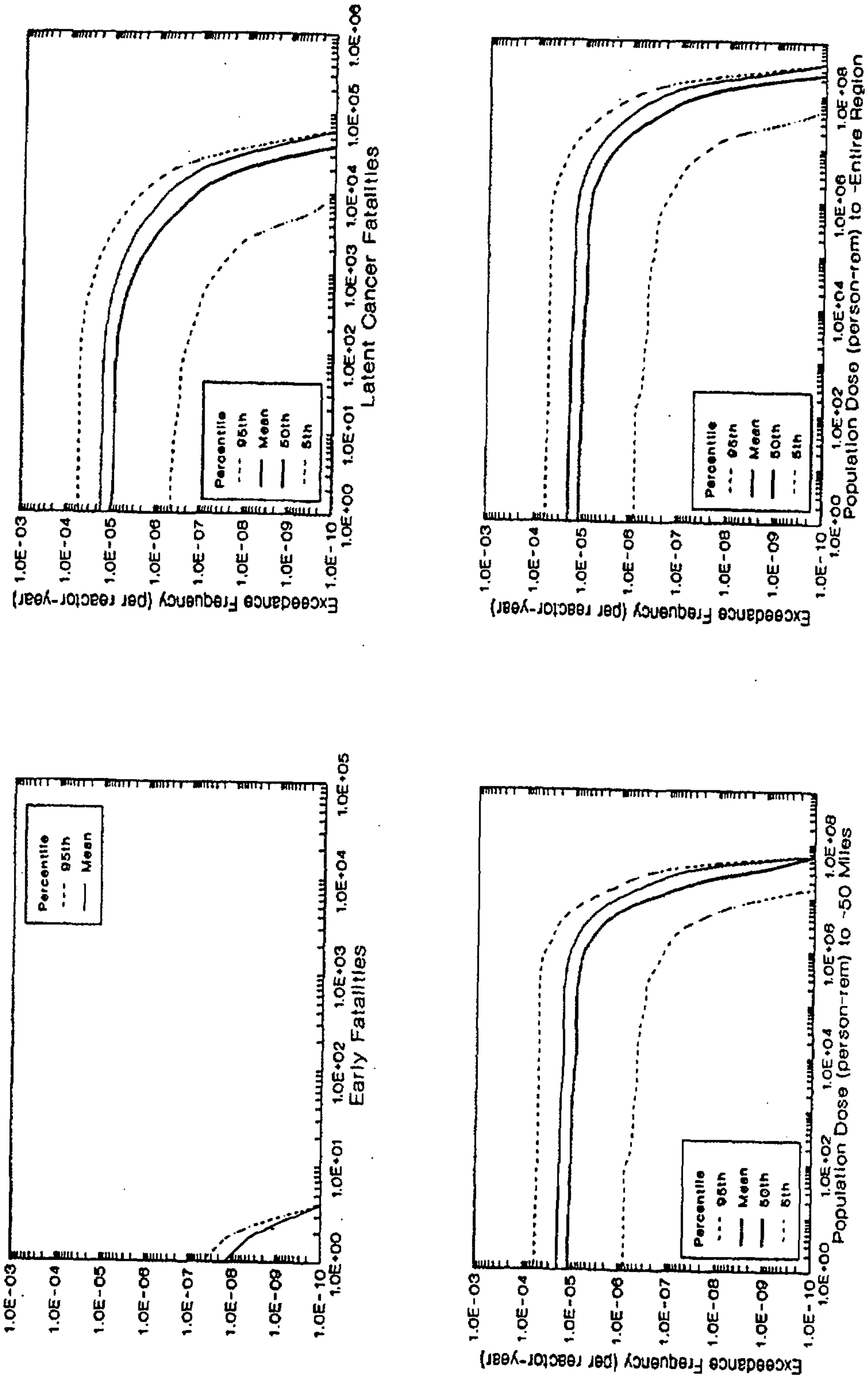
A detailed description of the results of the Peach Bottom risk is provided in Reference 4.2. For this summary report, results are provided for the following measures of public risk:

- Early fatality risk,
- Latent cancer fatality risk,
- Population dose within 50 miles of the site,
- Population dose within the entire site region,
- Individual early fatality risk in the population within 1 mile of the Peach Bottom exclusion area boundary, and
- Individual latent cancer fatality risk in the population within 10 miles of the site.



Note: As discussed in Reference 4.4, 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 4.9 Frequency distributions of offsite consequence measures at Peach Bottom (internal initiators).



Note: As discussed in Reference 4.4, 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 4.10 Frequency distributions of offsite consequence measures at Peach Bottom (fire initiators).

The first four of the above measures are commonly used measures in nuclear power plant risk studies. The last two are those used to compare with the NRC safety goals (Ref. 4.10).

4.6.1.1 Internally Initiated Accident Sequences

The results of the risk studies using the above measures are shown in Figures 4.11 through 4.13. The figures display the variabilities in mean risks estimated from the meteorology-averaged conditional mean values of the consequence measures. For the first two measures, the results of the first risk study of Peach Bottom, the Reactor Safety Study (Ref. 4.3), are also provided. As may be seen, the early fatality risk from Peach Bottom is estimated to be very low. Latent cancer fatality risks are lower than those of the Reactor Safety Study. The risks of population dose and individual early fatality risk are also very low, and the individual latent cancer fatality risk is orders of magnitude lower than the NRC safety goals. These comparisons are discussed in more detail in Chapter 12.

The risk results shown in Figure 4.11 have been analyzed to determine the relative contributions of plant damage states and accident progression bins to mean risk. The results of this analysis are provided in Figures 4.14 and 4.15. As can be seen from these figures, and from the supporting document (Ref. 4.2), the major contributors to both early and latent cancer fatality risks are from station blackout (SBO) and anticipated transients without scram (ATWS). The dominant accident progression bins are early containment failure and drywell failure caused by drywell meltthrough and loads at vessel breach (due to direct containment heating, steam blowdown, or quasistatic pressure from steam explosion).

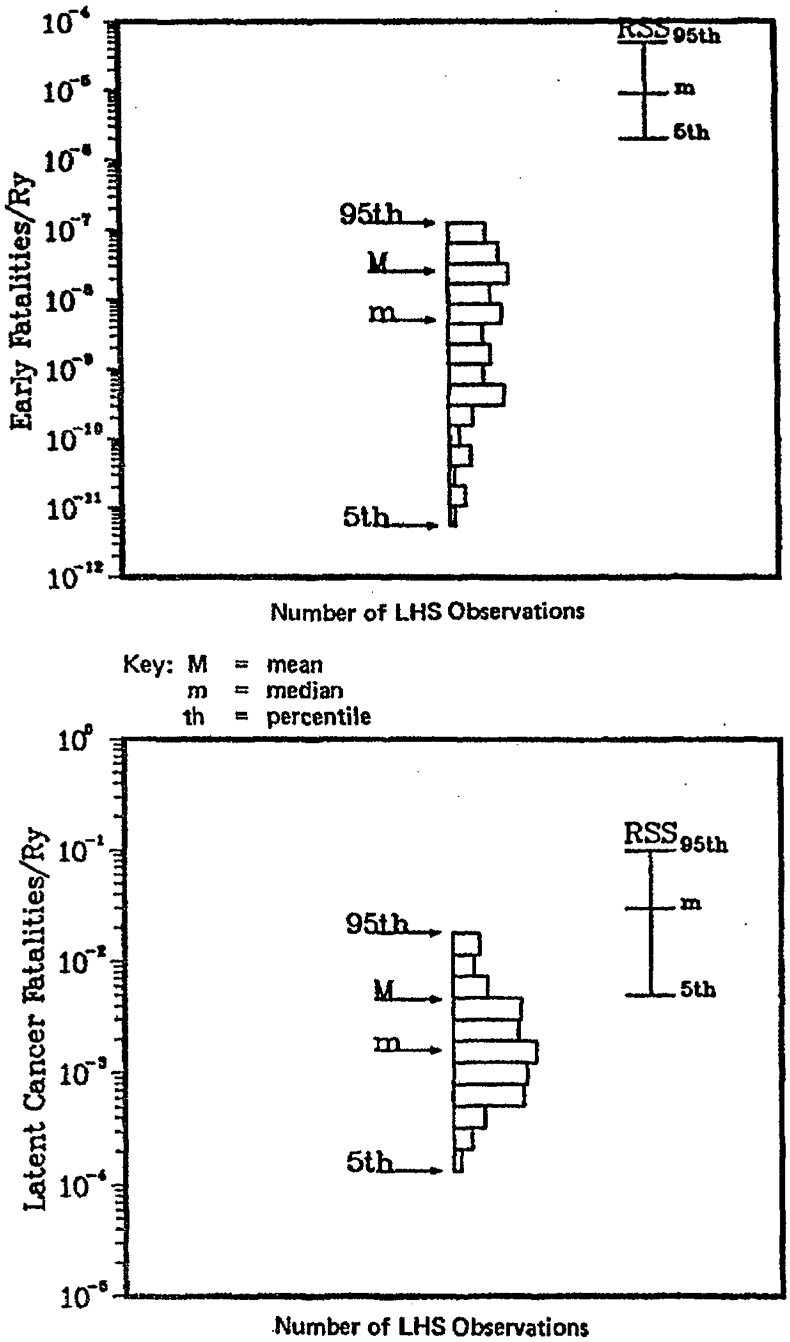
4.6.1.2 Externally Initiated Accident Sequences

As discussed in Section 4.2.1.2, the Peach Bottom plant has been analyzed for two externally initiated accidents: earthquakes and fire. The fire risk analysis has been performed through the estimates for consequences and risk measures, whereas, as explained in Chapter 2, the seismic analysis has been conducted up to containment performance. Sensitivity analyses of seismic risk at Peach Bottom are provided in Reference 4.2.

Results of fire risk analysis (variabilities in mean risks estimated from the meteorology-averaged conditional mean values of the consequence measures) of Peach Bottom are shown in Figures 4.16 through 4.18 for early fatality, latent cancer fatality, population dose (within 50 miles of the site and within the entire site region), and individual early and latent cancer fatality risks. Major contributions to early and latent cancer fatality risks are shown in Figure 4.19. As can be seen, early and latent cancer fatality risks for fire at Peach Bottom are dominated by early containment failure and drywell failure caused by drywell meltthrough and loads at vessel breach. Other risk measures are slightly higher than those for internally initiated events but well below NRC safety goals.

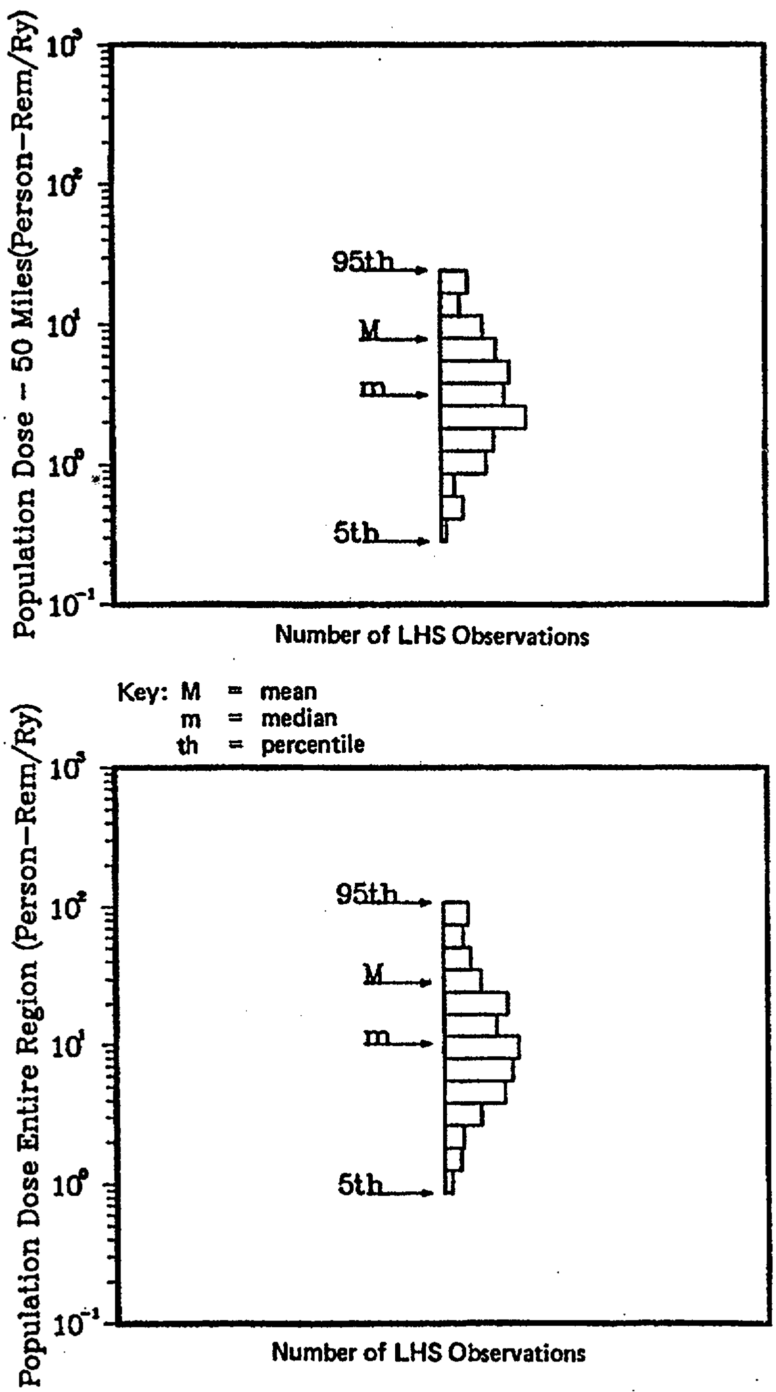
4.6.2 Important Plant Characteristics (Risk)

The risk from the internal events are driven by long-term station blackout (SBO) and anticipated transients without scram (ATWS). The dominance of these two plant damage states can be attributed to both general BWR characteristics and plant-specific design. BWRs in general have more redundant systems that can inject into the reactor vessel than PWRs and can readily go to low pressure and use their low-pressure injection systems. This means that the dominant plant damage states will be 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 only require a small number of systems to fail in order to reach core damage. The station blackout plant damage state satisfies the first of these requirements in that all systems ultimately depend upon ac power, and a loss of offsite power is a relatively high probability event. The total probability of losing ac power long enough to induce core damage is relatively high, although still low for a plant with Peach Bottom's design. The ATWS scenario is driven by the small number of systems that are needed to fail and the high stress upon the operators in these sequences.



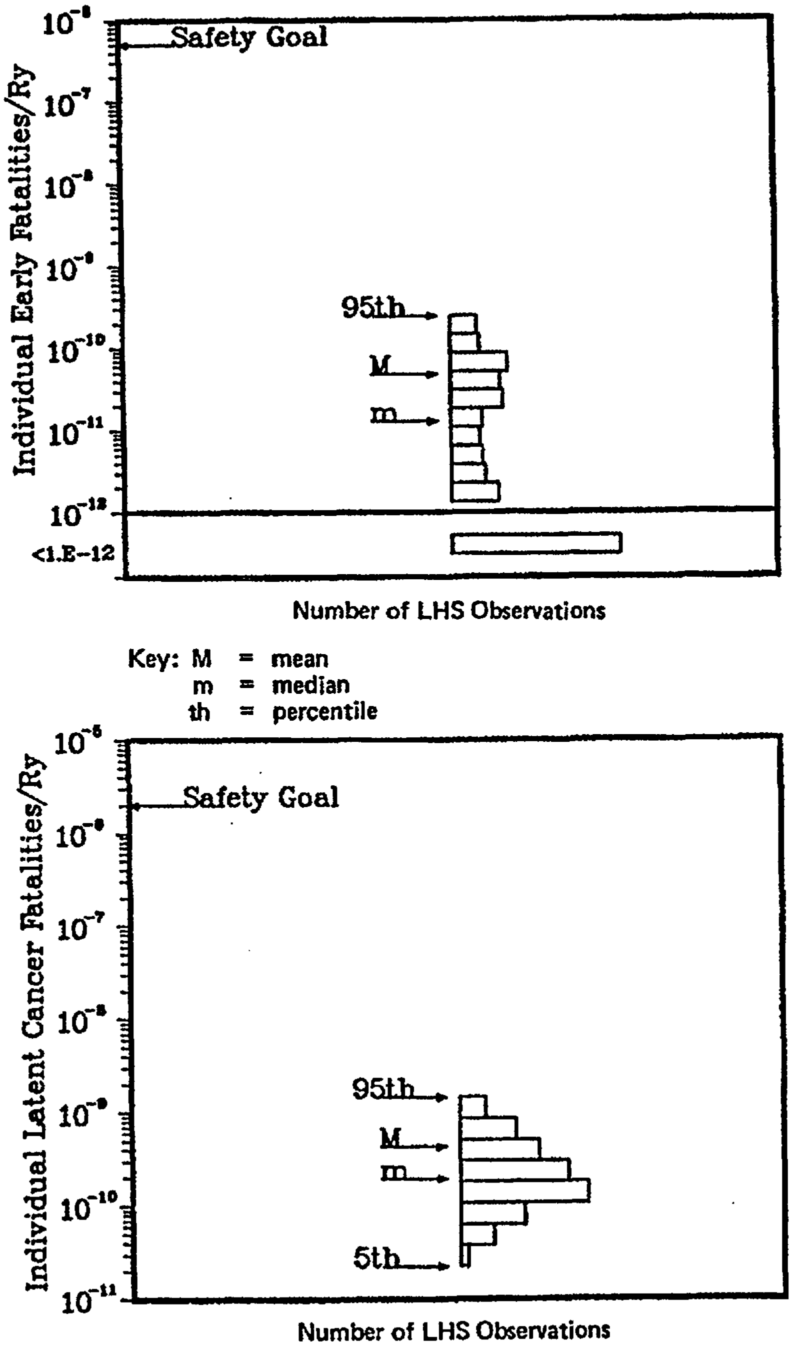
Note: As discussed in Reference 4.4, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 4.11 Early and latent cancer fatality risks at Peach Bottom (internal initiators).



Note: As discussed in Reference 4.4, 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 4.12 Population dose risks at Peach Bottom (internal initiators).



Note: As discussed in Reference 4.4, 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 4.13 Individual early and latent cancer fatality risks at Peach Bottom (internal initiators).

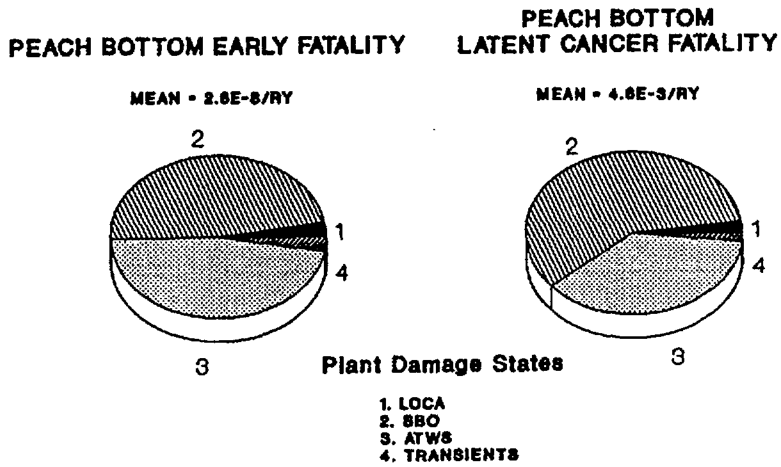


Figure 4.14 Major contributors (plant damage states) to mean early and latent cancer fatality risks at Peach Bottom (internal initiators).

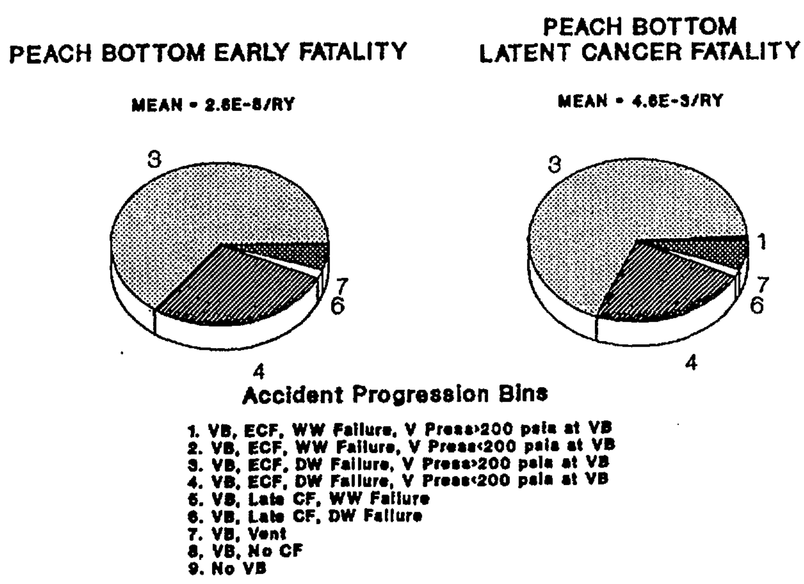
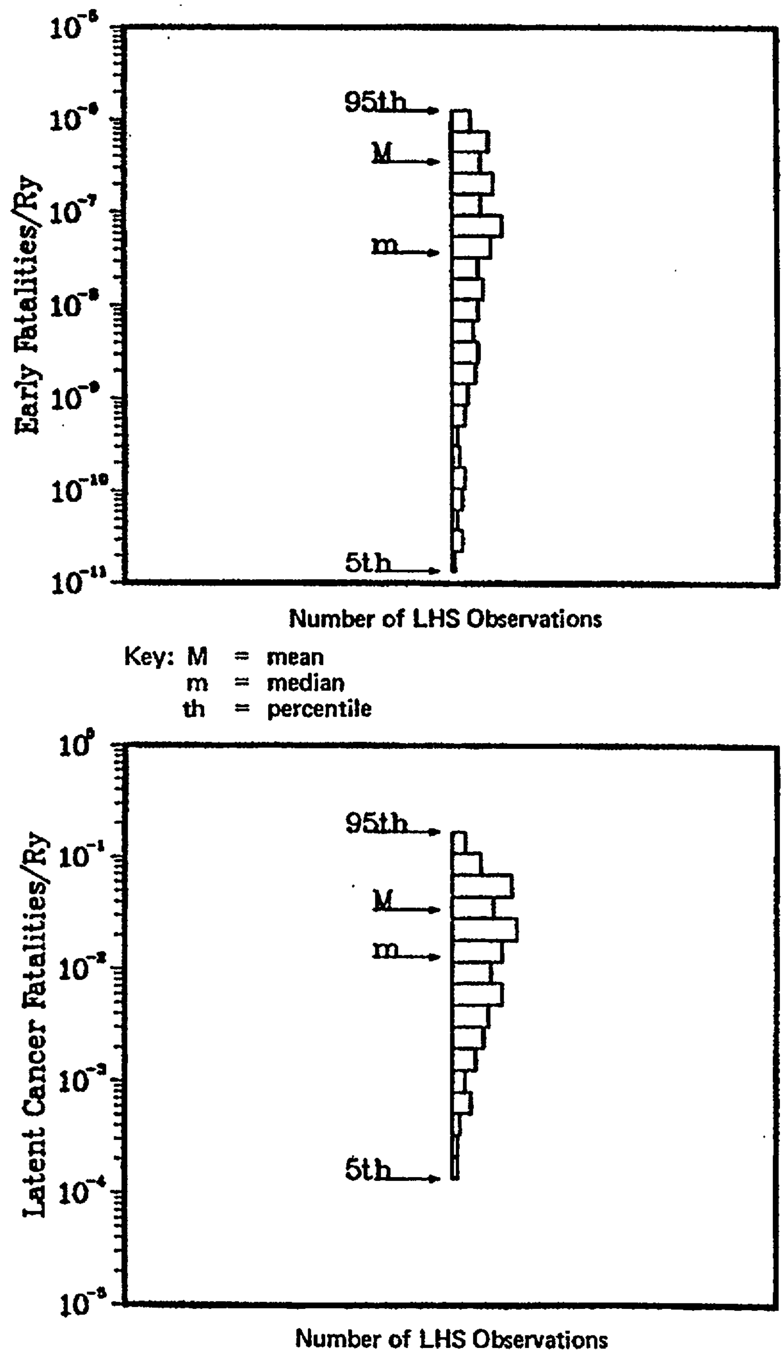
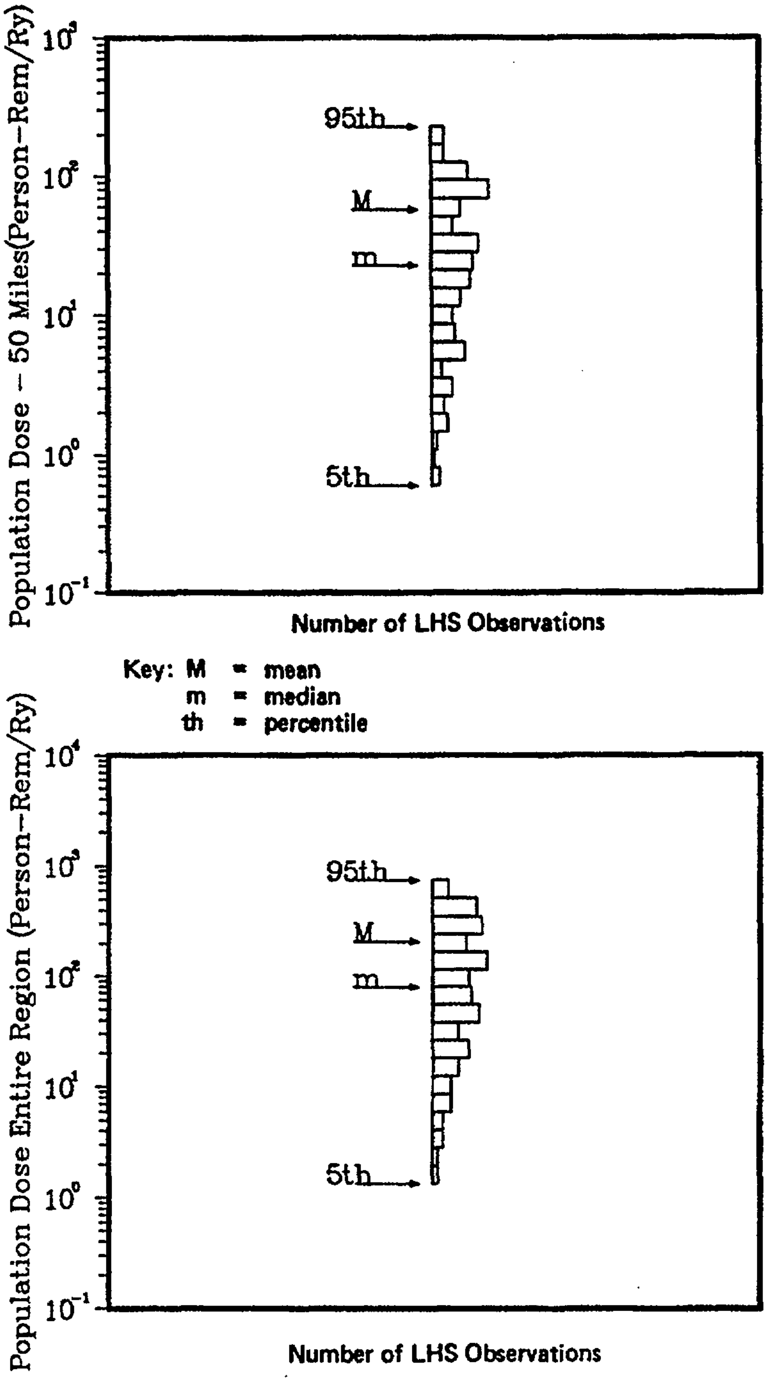


Figure 4.15 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Peach Bottom (internal initiators).



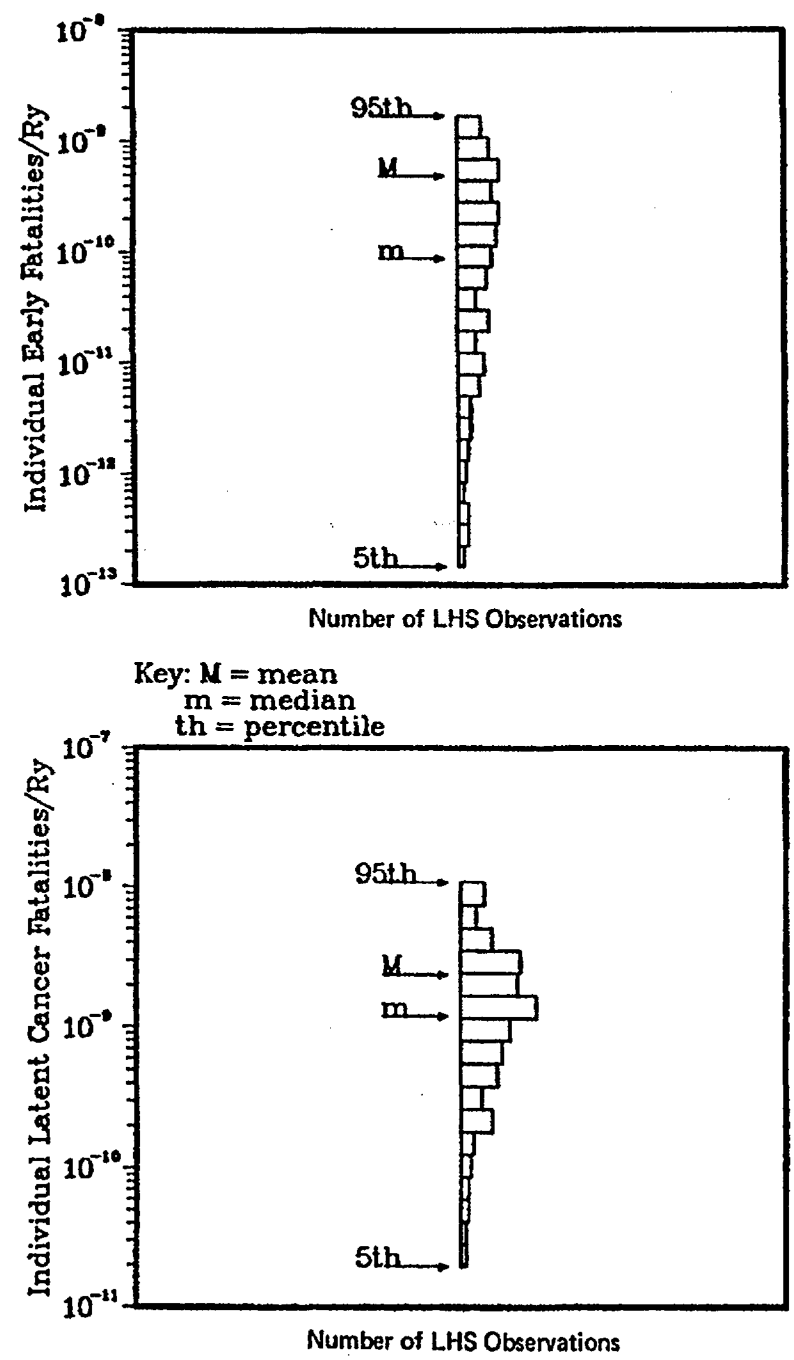
Note: As discussed in Reference 4.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 4.16 Early and latent cancer fatality risks at Peach Bottom (fire initiators).



Note: As discussed in Reference 4.4, 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 4.17 Population dose risks at Peach Bottom (fire initiators).



Note: As discussed in Reference 4.4, 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 4.18 Individual early and latent cancer fatality risks at Peach Bottom (fire initiators).

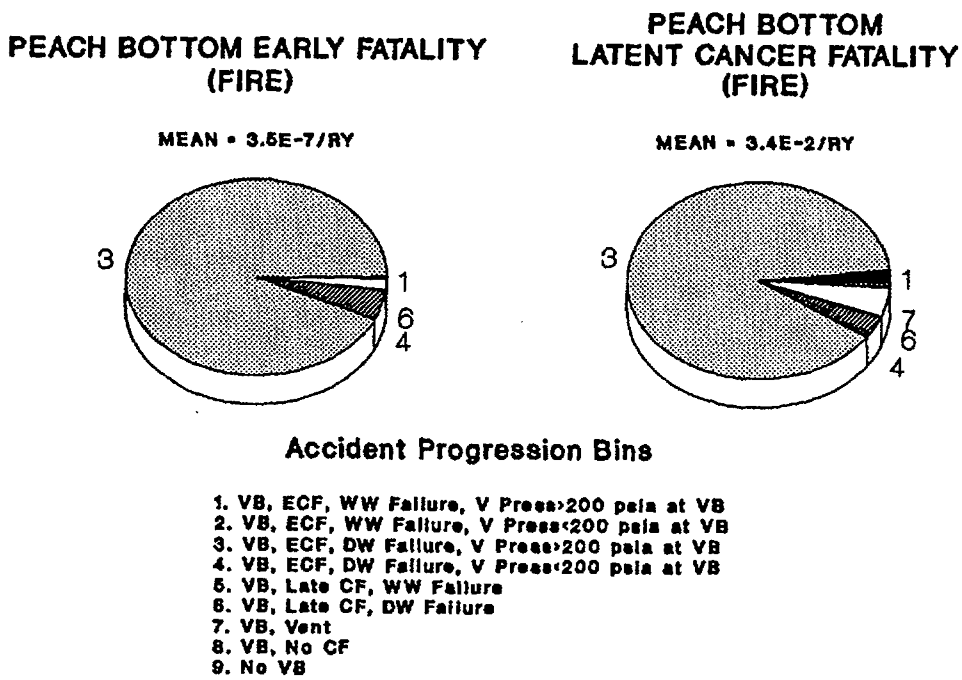


Figure 4.19 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Peach Bottom (fire initiators).

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4.2 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.*

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4.9 Philadelphia Electric Company, "Evacuation Time Estimates with the Plume Exposure Pathway Emergency Planning Zone for the Peach Bottom Atomic Power Station," Rev. 0, July 1982.

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

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

5. SEQUOYAH PLANT RESULTS

5.1 Summary Design Information

The Sequoyah Nuclear Power Plant is a two-unit site. Each unit, designed by Westinghouse Corporation, is a four-loop pressurized water reactor (PWR) rated at 1148 MWe and is housed in an ice condenser containment. The balance of plant systems were engineered and built by the utility, the Tennessee Valley Authority. Sequoyah 1 started commercial operation in 1981. Some important design features of the Sequoyah plant are described in Table 5.1. A general plant schematic is provided in Figure 5.1.

This chapter provides a summary of the results obtained in the detailed risk analyses underlying this report (Refs. 5.1 and 5.2). A discussion of perspectives with respect to these results is provided in Chapters 8 through 12.

5.2 Core Damage Frequency Estimates

5.2.1 Summary of Core Damage Frequency Estimates

The core damage frequency and risk analyses performed for this study considered accidents initiated only by internal events (Ref. 5.1); no external-event analyses were performed. The core damage frequency results obtained are provided in tabular form in Table 5.2 and in graphical form, displayed as a histogram, in Figure 5.2 (Section 2.2.2 discusses histogram development). This study calculated a total median core damage frequency from internal events of 3.7E−5 per year.

5.2.1.1 Internally Initiated Accident Sequences

Twenty-three individual accident sequences were identified as important to the core damage frequency estimates for Sequoyah. A detailed description of these accident sequences is provided in Reference 5.1. For the purpose of discussion here, the accident sequences have been grouped into five summary plant damage states. These are:

- Station blackout,
- Loss-of-coolant accidents (LOCAs),
- Anticipated transients without scram (ATWS),
- Transients other than station blackout and ATWS, and
- Interfacing-system LOCA and steam generator tube rupture (bypass accidents).

The relative contributions of these groups to the total mean core damage frequency at Sequoyah is shown in Figure 5.3. It is seen that loss-of-coolant accidents as a group are the largest contributors to core damage frequency. Within the general class of loss-of-coolant accidents, the most probable combinations of failures are:

- Intermediate (2" < D < 6"), small (1/2 < D < 2"), and very small (D < 1/2") size LOCAs in the reactor coolant system piping followed by failure of high-pressure or low-pressure emergency coolant recirculation from the containment sump. Coolant recirculation from the containment sump can fail because of valve failures, pump failures, plugging of drains or strainers, or operator failure to correctly reconfigure the emergency core cooling system (ECCS) equipment for the recirculation mode of operation.

Station blackout sequences as a group are the second largest contributor to core damage frequency. Within this group, the most probable combinations of failures are:

- Station blackout with failure of the auxiliary feedwater (AFW) system. Core uncovery is caused by failure of the AFW system to provide steam generator feed flow, thus causing gradual heatup and boiloff of reactor coolant. Station blackout also results in the unavailability of the high-pressure injection systems for feed and bleed. The dominant contributors to this sequence are the station blackout followed by initial turbine-driven AFW pump unavailability due to mechanical failure or maintenance outage, or failure of the operator to open air-operated valves after depletion of the instrument air supply.
- Station blackout with initial AFW operation that fails at a later time because of battery depletion or station blackout, with reactor coolant pump (RCP) seal LOCA because of loss of all RCP seal cooling. Station blackout results in a loss of seal injection flow to the RCPs and a loss of component cooling water to the RCP thermal barriers. This condition results in vulnerability of the RCP seals to

Table 5.1 Summary of design features: Sequoyah Unit 1.

1. Coolant Injection System	a. Charging system provides safety injection flow, emergency boration, feed and bleed cooling, and normal seal injection flow to the RCPs,* with 2 centrifugal pumps. b. RHR system provides low-pressure emergency coolant injection and recirculation following LOCA, with 2 trains and 2 pumps. c. Safety injection system provides high head safety injection and feed and bleed cooling, with 2 trains and 2 pumps.
2. Steam Generator Heat Removal Systems	a. Power conversion system. b. Auxiliary feedwater system, with 3 trains and 3 pumps (2 MDPs, 1 TDP).*
3. Reactivity Control Systems	a. Control rods. b. Chemical and volume control systems.
4. Key Support Systems	a. dc power, with 2-hour station batteries. b. Emergency ac power, with 2 diesel generators for each unit, each diesel generator dedicated to a 6.9 kV emergency bus (these buses can be crosstied to each other via a shutdown utility bus). c. Component cooling water provides cooling water to RCP* thermal barriers and selected ECCS equipment, with 5 pumps and 3 heat exchangers for both Units 1 and 2. d. Service water system, with 8 self-cooled pumps for both Units 1 and 2.
5. Containment Structure	a. Ice condenser. b. 1.2 million cubic feet. c. 10.8 psig design pressure.
6. Containment Systems	a. Spray system provides containment pressure-suppression during the injection phase following a LOCA and also provides containment heat removal during the recirculation phase following a LOCA. b. System of igniters installed to burn hydrogen. c. Air-return fans to circulate atmosphere through the ice condenser and keep containment atmosphere well mixed.

*MDP: Motor-Driven Pump
TDP: Turbine-Driven Pump
RCP: Reactor Coolant Pump

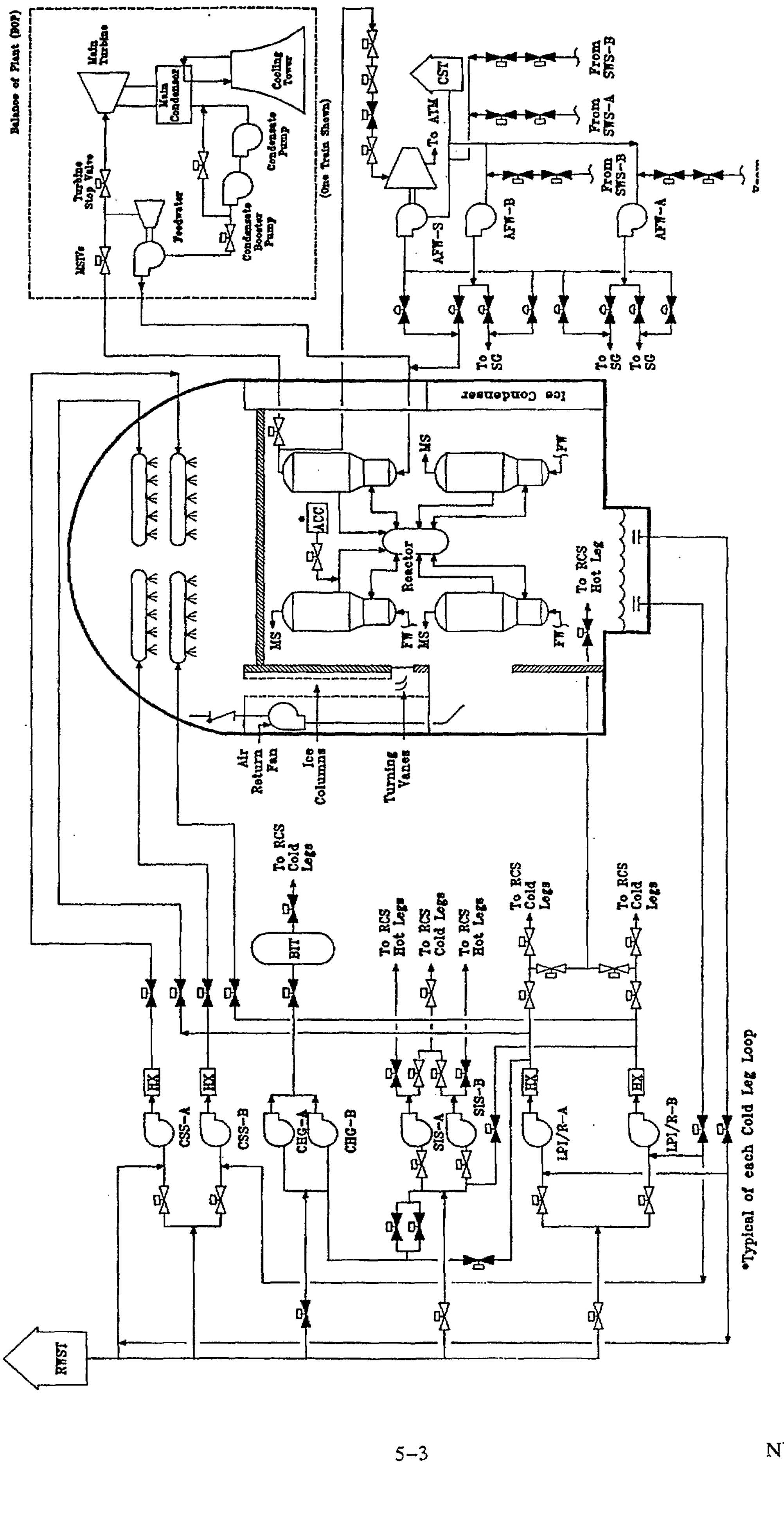
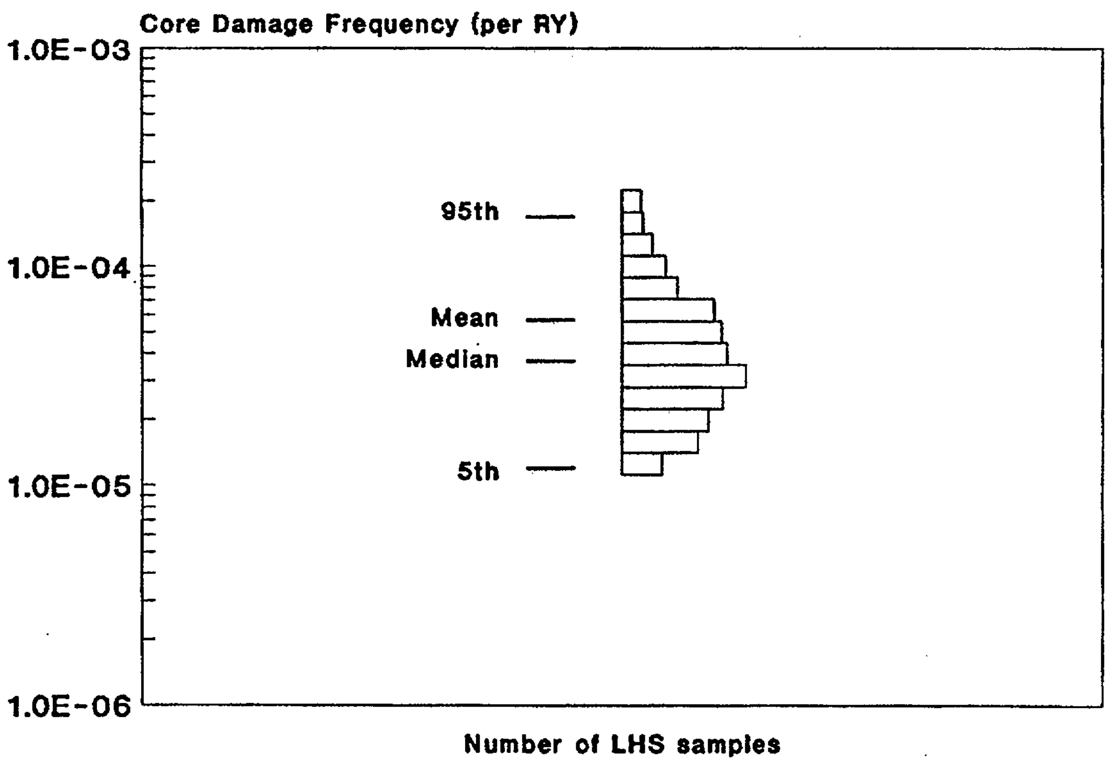


Figure 5.1 Sequoyah plant schematic.

Table 5.2 Summary of core damage frequency results: Sequoyah.*

	5%	Median	Mean	95%
Internal Events	1.2E-5	3.7E-5	5.7E-5	1.8E-4
Station Blackout				
Short Term	4.2E-7	3.8E-6	9.6E-6	3.6E-5
Long Term	1.0E-7	1.4E-6	5.0E-6	1.7E-5
ATWS	4.3E-8	5.3E-7	1.9E-6	7.5E-6
Transient	2.5E-7	1.1E-6	2.6E-6	7.2E-6
LOCA	4.4E-6	1.8E-5	3.6E-5	1.2E-4
Interfacing LOCA	1.5E-11	2.0E-8	6.5E-7	2.1E-6
SGTR	2.4E-8	4.1E-7	1.7E-6	7.1E-6

*As discussed in Reference 5.3, 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).



Note: As discussed in Reference 5.3, 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 5.2 Internal core damage frequency results at Sequoyah.

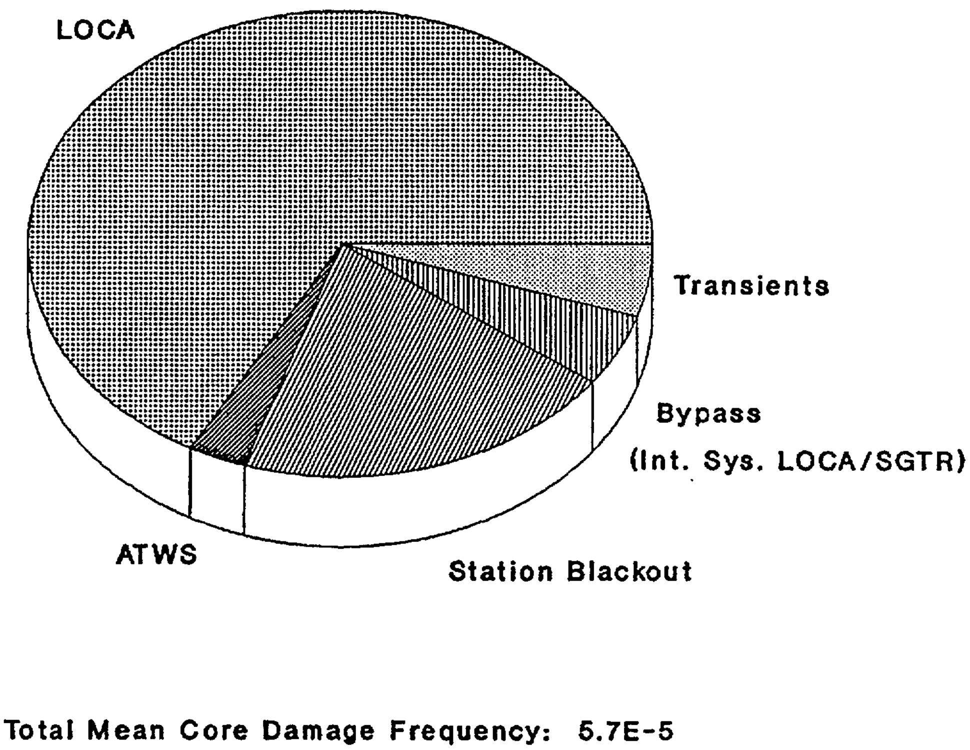


Figure 5.3 Contributors to mean core damage frequency from internal events at Sequoyah.

failure. The failure to restore ac power and safety injection flow following any seal LOCA leads to core uncovery. The time to core uncovery following onset of a seal LOCA is a function of the leak rate and whether or not the operator takes action to depressurize the reactor coolant system.

Within the general group of containment bypass accidents, the more probable combinations of failure are:

- Steam generator tube rupture, followed by failure to depressurize the reactor coolant system (RCS). Subsequent failure to depressurize the RCS in the long term and thus limit RCS leakage leads to continued blowdown through the steam generator and eventual core uncovery. An important event in this sequence is the initial failure of the operator to depressurize within 45 minutes after the tube rupture. This leads to a relief valve demand in the secondary cooling system. The steam generator safety valve will be demanded if the power-operated relief valve is blocked. Subsequent failure of the PORV or safety valve to reclose leads to direct loss of RCS inventory to the atmosphere. Failure of subsequent efforts to recover the sequence by RCS depressurization or closure of the PORV or safety valve leads to refueling water storage tank inventory depletion and eventual core uncovery.

- Failure of RCS pressure isolation leading to LOCAs in systems interfacing with the reactor coolant system (by overpressurization of low-pressure piping in the interfacing system). These sequences comprise 2 percent of the total core damage frequency but are important contributors to risk because they create a direct release path to the environment. These accidents are of special interest because they prevent ECCS operation in the recirculation mode and lead to containment bypass.

5.2.2 Important Plant Characteristics (Core Damage Frequency)

Characteristics of the Sequoyah plant design and operation that have been found to be important in the analysis of core damage frequency include:

1. **Electric Power Crossconnects Between Units 1 and 2**

   The Sequoyah electric power system design includes the capability to crosstie the 6.9 kV emergency buses at Unit 1 and Unit 2 and includes the capability to energize dc battery boards at Unit 1 from the batteries at Unit 2. These crossties help reduce the frequency of station blackout at Unit 1 and significantly reduce the possibility of battery depletion as an important contributor for those station blackouts that are postulated to occur. The crossties reduce the station blackout core damage frequency by less than a factor of 2. As station blackout sequences only account for 20 percent of the total core damage frequency, the crossties reduce total core damage frequency by approximately 10 percent.

2. **Transfer to Emergency Core Cooling and Containment Spray System Recirculation Mode**

   The process for switching the emergency core cooling system and the containment spray system from the injection mode to the recirculation mode at Sequoyah involves a series of operator actions that must be accomplished in a relatively short time (~20 minutes) and are only partially automated. Therefore, operator action is required to maintain core cooling when switching over to the recirculation mode. Single operator errors during switchover from injection to recirculation following a small LOCA can lead directly to core uncovery. Recirculation failure can also result from common-cause failures affecting the entire emergency core cooling system and containment spray system. These failures include level sensor miscalibration for the refueling water storage tank and failure to remove the upper containment compartment drain plugs after refueling.

3. **Loss of Coolant from Interfacing-System LOCA**

   Interfacing-system LOCA results from failures of any one of the four pairs of series check valves used to isolate the high-pressure RCS from the low-pressure injection system. The resultant flow into the low-pressure system is assumed to result in rupture of the low-pressure piping or components outside the containment boundary. Although core inventory makeup by the high-pressure injection system is initially available, the inability to switch to the recirculation mode would eventually lead to core damage. Because of the location of the postulated LOCA, all containment safeguards are bypassed.

   The failure scenarios of interest are those that produce a sudden large backleakage from the RCS that cannot be accommodated by relief valves in the low-pressure systems. Interfacing-system LOCA could therefore occur in two ways:

   a. Random or dependent rupture of valve internals on both valves. Rupture of the upstream valve would go undetected until rupture of the second valve occurred, and

   b. Rupture of the downstream valve combined with the failure of the upstream valve to be closed on demand. This scenario has an extremely low probability at Sequoyah because the check valve testing procedures require leak rate testing after each valve use.

   If an interfacing-system LOCA should occur, a potential recovery action was identified and considered in the analysis in which the operator may be able to isolate the interfacing-system LOCA by closing the appropriate low-pressure injection cold leg isolation valve.

4. **Diesel Generators**

   Sequoyah is a two-unit site with four diesel generator units. Each diesel is dedicated to a particular (6.9 kV) emergency bus at one of the units. Each diesel generator can only be connected to its dedicated emergency bus. However, the 6.9 kV buses can be crosstied to each other through the use of the shutdown utility bus, thus providing an indirect way to crosstie diesels and emergency buses. The diesel generators have dedicated batteries for starting and can be loaded on the emergency buses manually or with alternative power supplies. Emergency ac power is therefore not as susceptible to failures of the station batteries as at those plants where station batteries are used for diesel startup.

5. **Containment Design**

The ice condenser containment design is important to estimates of core damage frequency because of the spray actuation setpoints. The relatively low-pressure setpoints result in spray actuation for a significant percentage of small LOCAs. The operation of the sprays will deplete the refueling water storage tank (RWST) in approximately 20 minutes, thus requiring fast operator intervention to switch over to recirculation mode. The reduced time available for operator action results in an increased human error rate for recirculation alignment associated with this time interval.

5.2.3 Important Operator Actions

Several operator actions are very important in preventing core uncovery. These actions are discussed in this section with respect to the accident sequence in which they occur.

- Switchover to ECCS recirculation in a small LOCA

  There are four major operator actions during recirculation switchover:

  - Switchover of high-pressure emergency core cooling system (ECCS) from injection to recirculation.
  - Isolation of ECCS suction from RWST.
  - Switchover of containment spray system (CSS) from injection to recirculation, including isolation of suction from the RWST.
  - Valving in component cooling water (CCW) to the residual heat removal (RHR) heat exchangers.

- Control of containment sprays during small LOCAs

  Virtually all small LOCAs will result in automatic containment spray actuation. If the operator does not control sprays early during a small LOCA, the RWST level will decrease and switchover to recirculation will be required.

  All actions are performed in the main control room at one location. The time for diagnosis is relatively short (~20 minutes) for determining if the event is actually a LOCA and anticipating whether high-pressure recirculation will be needed when the low RWST level alarm is actuated.

- Feed and bleed cooling

  For accident sequences in which main and auxiliary feedwater are unavailable, feed and bleed cooling can be used to remove decay heat from the core. The operator is instructed to initiate feed and bleed cooling if steam generator levels drop below 25 percent. This point is reached approximately 30 minutes after auxiliary feedwater (AFW) and main feedwater become unavailable.

- Anticipated transients without scram (ATWS)

  Five operator actions could potentially be required during an ATWS sequence, depending on the particular course of the sequence. These events are:

  - Manual reactor trip.
  - Trip turbine if not done automatically.
  - Start AFW if not started automatically.
  - Open block valve on power-operated relief valve (PORV) within 2 minutes if PORV is isolated previous to initiating event.
  - Emergency boration, if manual trip failed.

  Due to the fast-acting nature of an ATWS, all ATWS actions must be performed from memory.

- Steam generator tube rupture

  Steam generator tube rupture (SGTR) accident sequences are considered to begin with a double-ended rupture of a single steam generator tube. Very shortly thereafter, a safety injection signal will occur on low RCS pressure. The immediate concern for the operator, after identifying the event as an SGTR, is to identify and isolate the ruptured steam generator. There are three possible operator actions during an SGTR. These are:

  - Cool down and depressurize the RCS very shortly (~45 minutes) after the event in order to prevent lifting the relief valves on the affected steam generator;

- Restore the main feedwater flow in the event of a loss of auxiliary feed flow; and

- Isolate the steam generator that contains the ruptured tube.

- Interfacing-system LOCA recovery action

  The two RHR trains are physically isolated from each other and are provided with system isolation capability. To recover from an interfacing-system LOCA in the RHR system and to continue core cooling, the break must first be isolated and the reactor coolant system refilled. Since the RHR valves are not designed to close against the pressure differentials present during the blowdown, isolation of the affected loop and operation of the unaffected loop must be accomplished following blowdown. The RHR valves can be closed from the control room. No credit for local action is given because of the steam environment following the blowdown.

5.2.4 Important Individual Events and Uncertainties (Core Damage Frequency)

As discussed in Chapter 2, the process of developing a probabilistic model of a nuclear power plant involves the combination of many individual events (initiators, hardware failures, operator errors, etc.) into accident sequences and eventually into an estimate of the total frequency of core damage. After development, such a model can also be used to assess the importance of the individual events. The detailed studies underlying this report have been analyzed using several event importance measures. The results of the analyses using two measures, "risk reduction" and "uncertainty" importance, are summarized below.

- Risk (core damage frequency) reduction importance measure (internal events)

  The risk-reduction importance measure is used to assess the change in core damage frequency as a result of setting the probability of an individual event to zero. Using this measure, the following individual events were found to cause the greatest reduction in core damage frequency if their probabilities were set to zero:

  - Very small LOCA initiating event. The core damage frequency will be reduced by approximately 38 percent.

  - Operator fails to control sprays during a small LOCA. The core damage frequency will be reduced by approximately 37 percent.

  - Loss of offsite power initiating event. The core damage frequency will be reduced by approximately 21 percent.

  - Operator failure to properly align high-pressure recirculation. The core damage frequency will be reduced by approximately 15 to 20 percent.

  - Failure to recover diesel generators within 1 hour. The core damage frequency will be reduced by approximately 14 percent.

  - Failure to recover ac power within 1 hour. The core damage frequency will be reduced by approximately 13 percent.

  - Intermediate LOCA initiating events. The core damage frequency will be reduced by approximately 12 percent.

  - Small LOCA initiating events. The core damage frequency will be reduced by approximately 13 percent.

- Uncertainty importance measure (internal events)

  A second importance measure used to evaluate the core damage frequency analysis results is the uncertainty importance measure. For this measure, the relative contribution of the uncertainty of individual events to the uncertainty in total core damage frequency is calculated. Using this measure, the largest contributors to uncertainty in the results are the human error probabilities for failure to reconfigure the ECCS for high-pressure recirculation. All other events contribute relatively little to the uncertainty in overall core damage frequency.

5.3 Containment Performance Analysis

5.3.1 Results of Containment Performance Analysis

The Sequoyah primary containment consists of a pressure-suppression containment system, i.e., ice condenser, which houses the reactor pressure vessel, reactor coolant system, and the steam generators for the secondary side steam supply system. The containment system is comprised of a steel vessel surrounded by a concrete shield building enclosing an annular space. The internal containment volume, which has a total capacity of 1.2 million cubic feet, is divided into two major compartments connected by the ice condenser system, with the reactor coolant system occupying the lower compartment. The ice condenser is essentially a cold storage ice-filled room 50 feet in height, bounded on one side by the steel containment wall. The design basis pressure for Sequoyah's ice condenser containment is 10.8 psig, whereas its estimated mean failure pressure is 65 psig. This low-pressure design combined with the relatively small free volume made hydrogen control a design basis consideration, i.e., recombiners, and also a major consideration with respect to containment integrity for severe accidents, i.e., igniters and air-return fans. Similar to other containment design analyses for this study, the estimate of where and when Sequoyah's containment will fail relied heavily on the use of expert judgment to interpret and supplement the limited data available (Ref. 5.4).

The potential for early containment failure has been of considerable concern for Sequoyah since the steel containment has such a low design pressure. The principal mechanisms threatening the containment are hydrogen combustion effects, overpressurization due to direct containment heating, failure of the wall by direct contact with molten core material, and isolation failures.

The results of the Sequoyah containment analysis are summarized in Figures 5.4 and 5.5. Figure 5.4 displays information in which the conditional probabilities of ten containment-related accident progression bins; e.g., VB-early CF (during CD), are presented for each of five plant damage states. This information indicates that, on a frequency-weighted average,* the mean conditional probability from internal events of (1) early containment failure due to effects such as hydrogen combustion, direct containment heating, and wall contact failure is 0.07, (2) late containment failure due primarily to basemat meltthrough is 0.21, (3) containment bypass is 0.06, and (4) probability of no containment failure or no vessel breach is 0.66. It should be noted, however, that the conditional probabilities of early containment failure for the loss of offsite power (LOSP) plant damage state are considerably higher than the averaged values, i.e., about 0.13 for LOSP sequences involving vessel breach and 0.17 when those LOSP sequences having no vessel breach are included. Figure 5.5 further develops the conditional probability distribution of early containment failure for each of the plant damage states, providing the estimated range of uncertainties in the containment failure predictions. Overall conclusions that can be drawn from this information are discussed in Chapter 9. However, it should be noted that Sequoyah's early containment failure probability depends heavily on the accuracy of our predictions of core arrest probability, direct containment heating, hydrogen combustion, and wall attack effects.

Additional discussions on containment performance (for all studied plants) are provided in Chapter 9.

5.3.2 Important Plant Characteristics (Containment Performance)

Characteristics of the Sequoyah design and operation that are important to containment performance include:

1. **Pressure-Suppression Design**

   The Sequoyah ice condenser suppression design can have a significant effect on certain accident sequence risk results. For example, the availability of ice in the ice condenser can reduce the risk significantly from events involving steam or direct containment heating threats to the containment. In contrast, its availability during some station blackout sequences can result in a potentially combustible hydrogen concentration at the exit of the ice bed. Further discussion of the ice condenser pressure-suppression system relative to other PWR dry containments is contained in Chapter 9.

2. **Hydrogen Ignition System**

   The Sequoyah hydrogen ignition system will significantly reduce the threat to containment from uncontrolled hydrogen combustion

*Each value in the column in Figure 5.4 labeled "All" is obtained by calculating the products of individual accident progression bin conditional probabilities for each plant damage state and the ratio of the frequency of that plant damage state to the total core damage frequency.

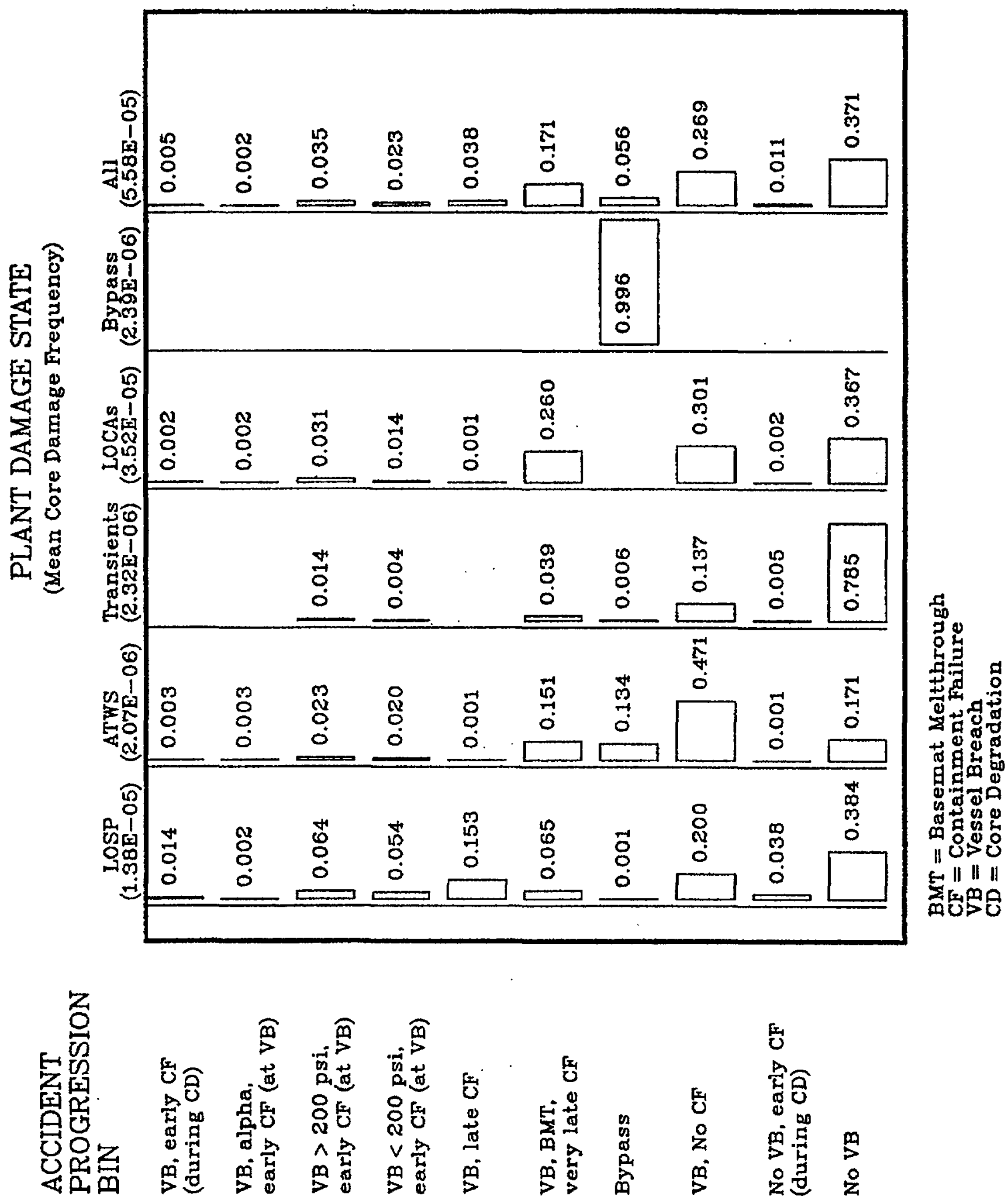


Figure 5.4 Conditional probability of accident progression bins at Sequoyah.

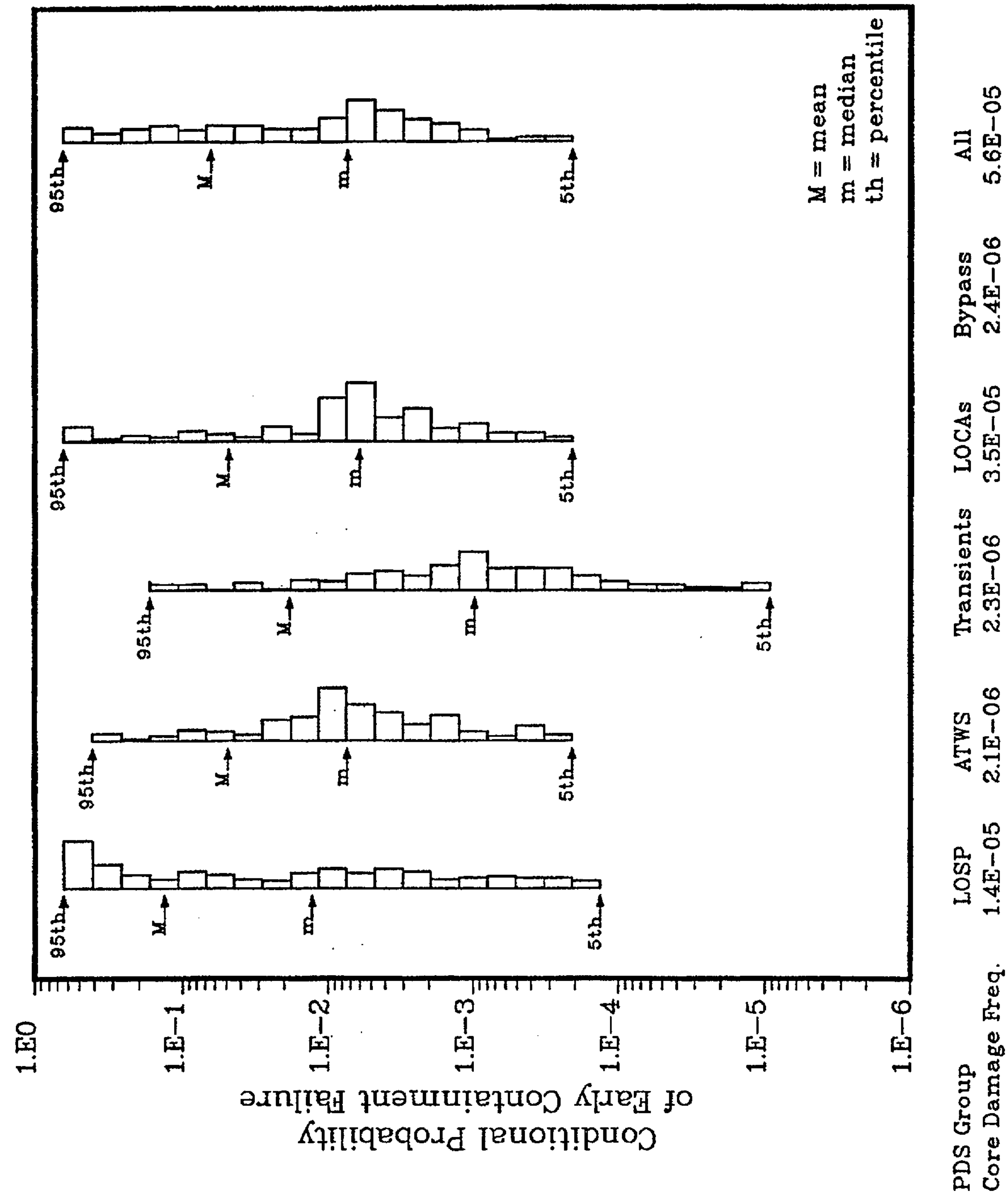


Figure 5.5 Conditional probability distributions for early containment failure at Sequoyah.

effects except for station blackout sequences. However, when power is recovered following a station blackout, if the igniters are turned on before the air-return fans have diluted the hydrogen concentration at or above the ice beds, the ignition could trigger a detonation or deflagration that could fail containment. These blackout sequences, however, represent a small fraction of the overall frequency of core damage.

3. **Lower Compartment Design**

   The design and construction of the seal table is such that if the reactor coolant system is at an elevated pressure upon vessel breach, the core debris is likely to get into the seal table room, which is directly in contact with the containment, and melt through the wall causing a break of containment. The design of the reactor cavity, however, does have the potential to cool the molten core debris and also mitigate the effects of potential direct containment heating events for those sequences where water is in the reactor cavity.

5.4 Source Term Analysis

5.4.1 Results of Source Term Analysis

The absolute frequencies of early containment failure from severe accident loads and of containment bypass are predicted to be similar for the Sequoyah plant (Ref. 5.2). Figure 5.6 illustrates the release fractions for an early containment failure accident progression bin. The mean values for the release of the volatile radionuclide groups are approximately 10 percent, indicative of an accident with the potential for causing early fatalities. The in-vessel releases in these accidents can be subject to decontamination by the ice bed or by containment sprays following release to the containment. The sprays require ac power and are, therefore, not available prior to power recovery in station blackout plant damage states. The decontamination factor of the ice bed is also affected by the unavailability of the recirculation fans during station blackout.

The location and mode of containment failure are particularly important for early containment failure accident progression bins. A substantial fraction of the early failures result in subsequent bypass of the ice bed. In particular, if the containment ruptures as the result of a sudden, high-pressure load, such as from hydrogen deflagration, the damage to the containment wall could be extensive and is likely to result in bypass.

In most accident sequences for Sequoyah, there is substantial water in the cavity that can either prevent core-concrete attack, if a coolable debris bed is formed, or mitigate the release of radionuclides during core-concrete attack by scrubbing in the overlaying water pool. As a result, a large release to the environment of the less volatile radionuclides that are released from fuel during core-concrete attack is unlikely for the Sequoyah plant.

In the station blackout plant damage state, containment failure can occur late in the accident as the result of hydrogen combustion following power recovery. Figure 5.7 illustrates the source terms for a late containment failure accident progression bin in which it is unlikely that water would be available to scrub the core-concrete releases. In this case, decontamination by the ice bed is important in mitigating the environmental release. As discussed previously, for very wide ranges of uncertainty covering many orders of magnitude, one or more high results can dominate the mean such that it falls above the 95th percentile.

5.4.2 Important Plant Characteristics (Source Term)

1. **Ice Condenser**

   In addition to condensing steam, the ice beds can trap radioactive aerosols and vapors in a severe accident. The extent of decontamination 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 single pass through the ice condenser with high steam fraction, the range of decontamination factor used in this study was from 1.3 to 35 with a median of 7 for the in-vessel release and less than half as effective for the core-concrete release. For the low steam fraction scenarios with a single pass through the ice beds, the lower bound was approximately 1.1, the upper bound 8, and the median 2. The values used for multiple passes through the ice bed when the containment is intact and the air-return fans are running are only slightly larger, with a median value of 3. Thus, the credit for ice bed retention is substantially less than the values used for the decontamination effectiveness of suppression pools in the BWRs.

2. **Cavity Configuration**

   The Sequoyah reactor cavity will be flooded if there is sufficient water on the containment floor to overflow into the cavity. If the contents of the refueling water storage tank are

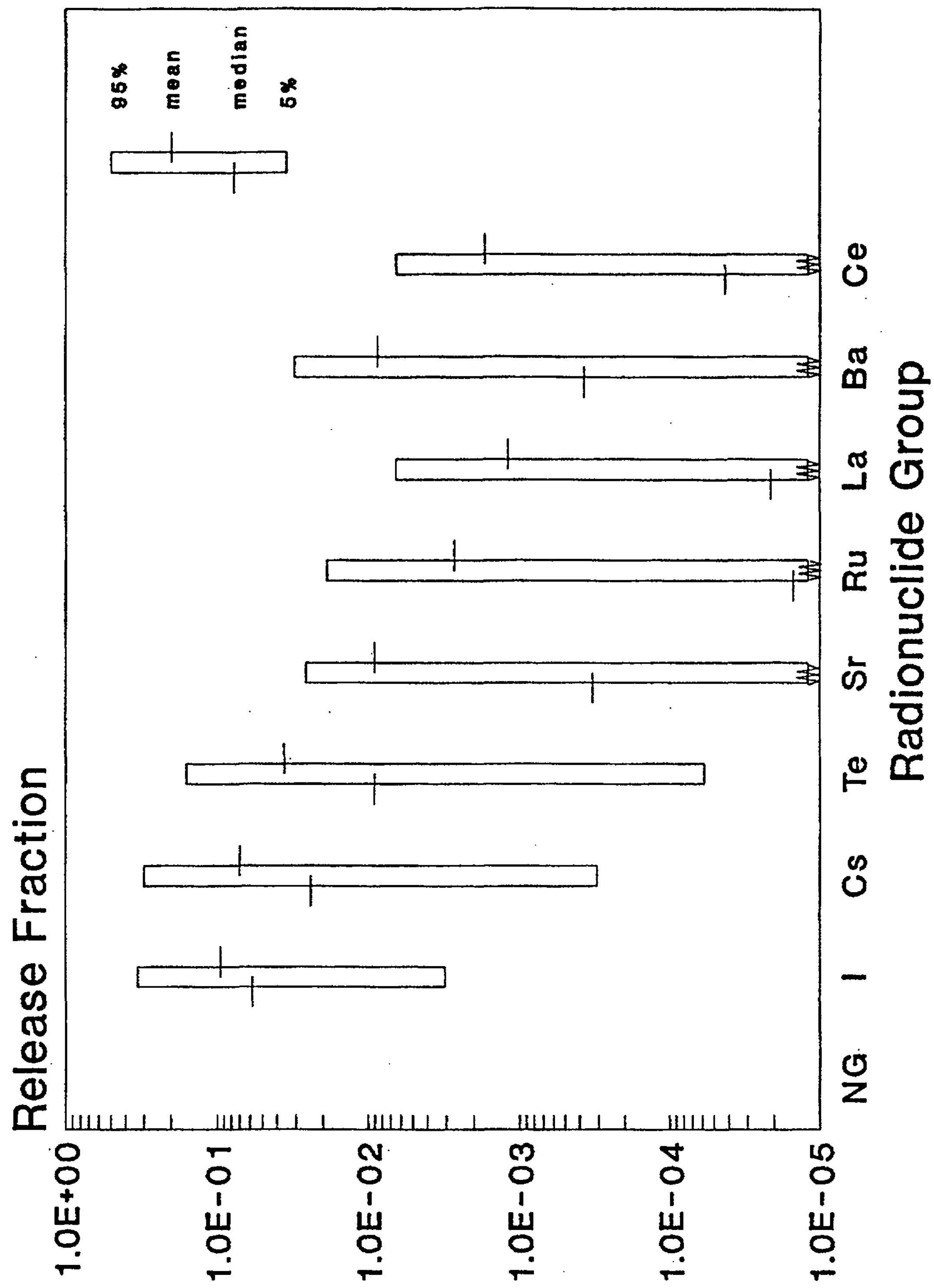


Figure 5.6 Source term distributions for early containment failure at Sequoyah.

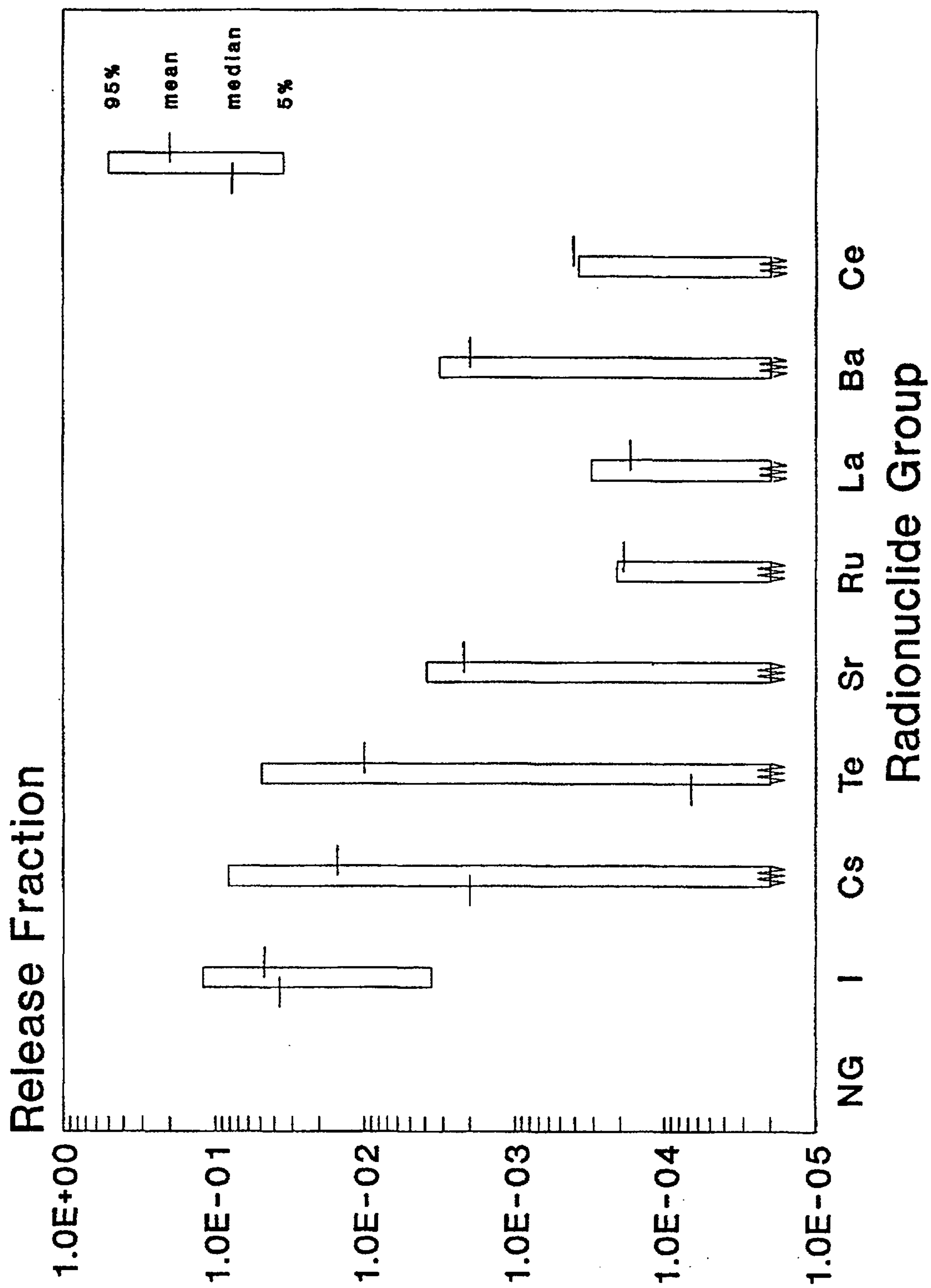


Figure 5.7 Source term distributions for late containment failure at Sequoyah.

discharged into the containment (e.g., by the spray system) and there is substantial ice melting, the water level in the cavity can be as high as 40 feet, extending to the level of the reactor coolant system hot legs. A decontamination factor for the deep water pool was used in the analyses, which ranged from approximately 4 to 9,000 with a median value of approximately 10 for the less volatile radionuclides released ex-vessel. If neither source of water to the containment is available, however, there will be no water in the cavity.

3. **Spray System**

   The Sequoyah containment has a spray system in the upper compartment to condense steam that bypasses the ice beds and for use after the ice has melted. As in the Surry plant, the spray system has the potential to dramatically reduce the airborne concentration of radioactive material if the containment remains intact for an extended period of time.

5.5 Offsite Consequence Results

Figure 5.8 displays the frequency distributions in the form of graphical plots of the complementary cumulative distribution functions (CCDFs) of four offsite consequence measures—early fatalities, latent cancer fatalities, and the 50-mile and entire site region population exposures (in person-rems). These CCDFs include contributions from all source terms associated with reactor accidents caused by internal initiating events. Four CCDFs, namely, the 5th percentile, 50th percentile (median), 95th percentile, and the mean CCDFs, are shown for each consequence measure.

Sequoyah plant-specific and site-specific parameters were used in the consequence analysis for these CCDFs. The plant-specific parameters included source terms and their frequencies, the licensed thermal power (3423 MWt) of the reactor, and the appropriate physical dimensions of the power plant building complex. The site-specific parameters included exclusion area radius (585 meters), meteorological data for 1 full year collected at the site meteorological tower, the site region population distribution based on the 1980 census data, topography (fraction of the area that is land—the remaining fraction is assumed to be water), land use, agricultural practice and productivity, and other economic data for up to 1,000 miles from the Sequoyah plant.

The consequence estimates displayed in these figures have incorporated the benefits of the following protective measures: (1) evacuation of 99.5 percent of the population within the 10-mile plume exposure pathway emergency planning zone (EPZ), (2) early relocation of the remaining population only from the heavily contaminated areas both within and outside the 10-mile EPZ, and (3) decontamination, temporary interdiction, or condemnation of land, property, and foods contaminated above acceptable levels.

The population density within the Sequoyah 10-mile EPZ is about 120 persons per square mile. The average delay time before evacuation (after a warning prior to radionuclide release) from the 10-mile EPZ and average effective evacuation speed used in the analyses were derived from information contained in a utility-sponsored Sequoyah evacuation time estimate study (Ref. 5.5) and the NRC requirements for emergency planning.

The results displayed in Figure 5.8 are discussed in Chapter 11.

5.6 Public Risk Estimates

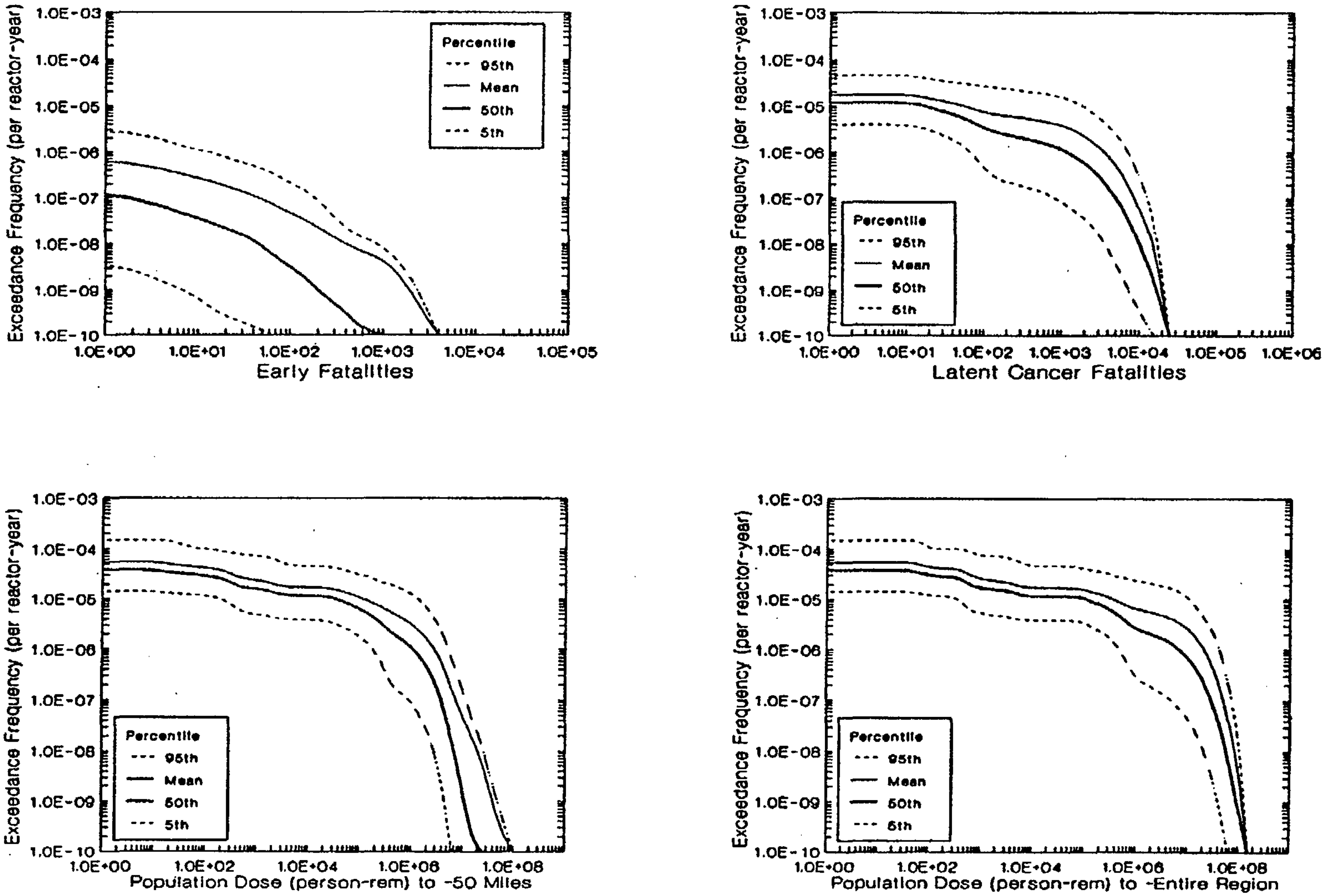
5.6.1 Results of Public Risk Estimates

A detailed description of the results of the Sequoyah risk is provided in Reference 5.2. For this summary report, results are provided for the following measures of public risk:

- Early fatality risk,
- Latent cancer fatality risk,
- Population dose within 50 miles of the site,
- Population dose within the entire site region,
- Individual early fatality risk in the population within 1 mile of the Sequoyah boundary, and
- Individual latent cancer fatality risk in the population within 10 miles of the Sequoyah site.

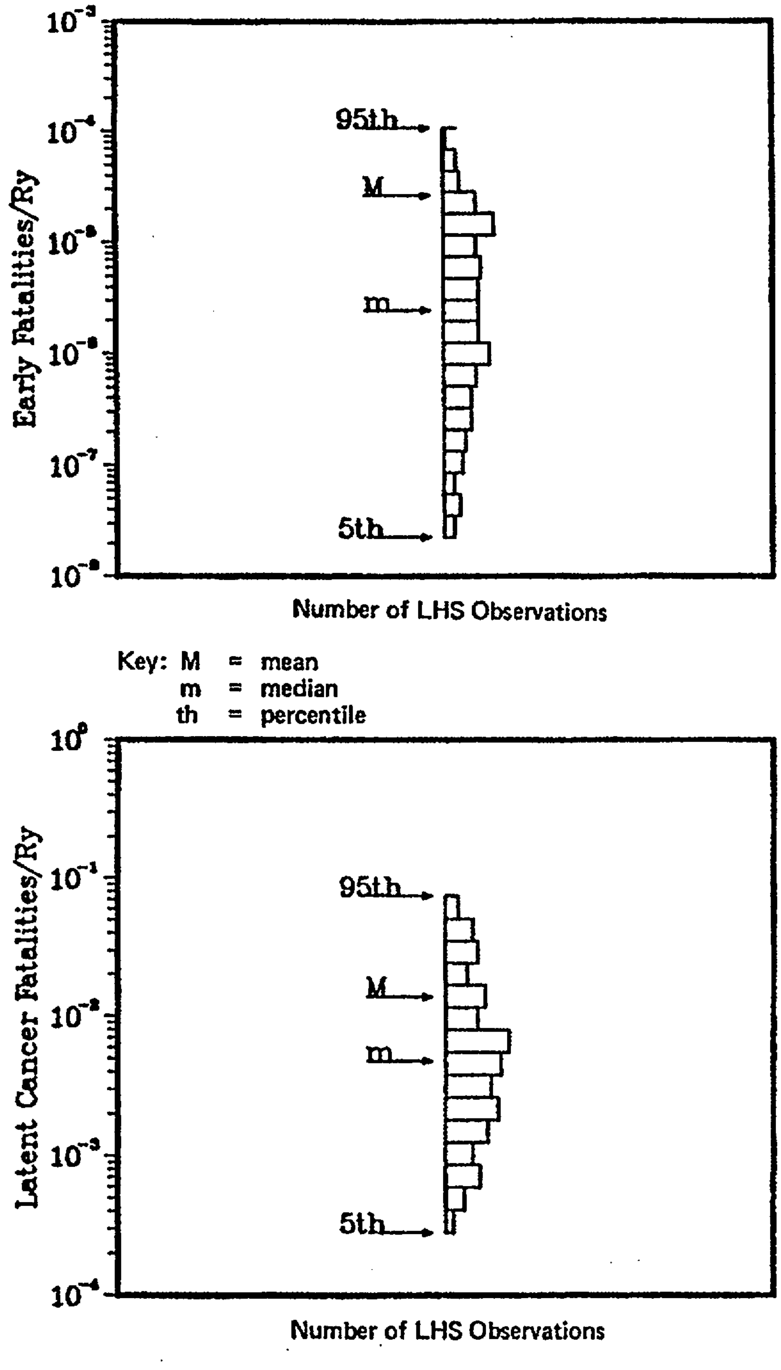
The first four of the above measures are commonly used measures in nuclear power plant risk studies. The last two are those used to compare with the NRC safety goals (Ref. 5.6).

The results of Sequoyah risk analysis using the above measures are shown in Figures 5.9 through 5.11. The figures display the variabilities in mean risks estimated from the meteorology-averaged mean values of the consequence measures. The



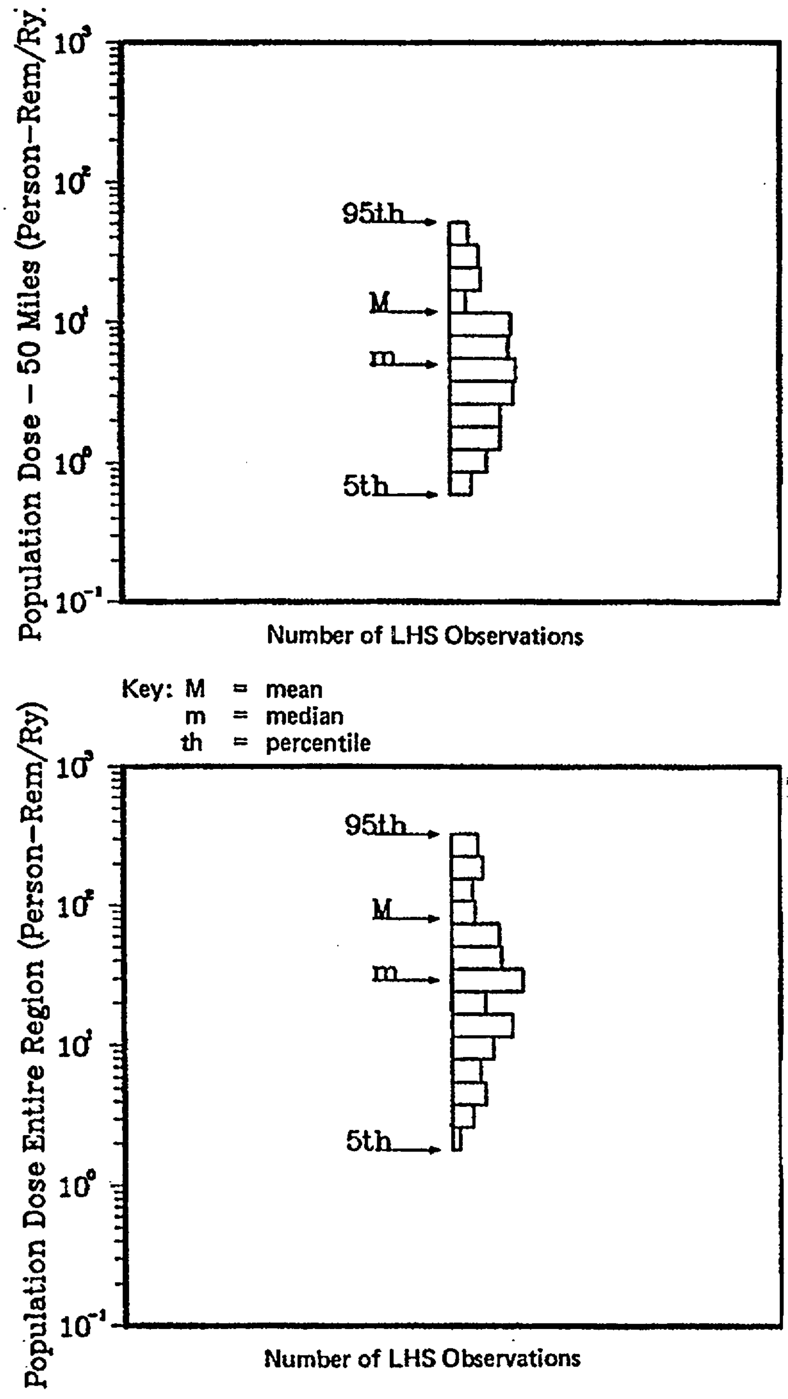
Note: As discussed in Reference 5.3, estimated consequences at frequencies at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 5.8 Frequency distributions of offsite consequence measures at Sequoyah (internal initiators).



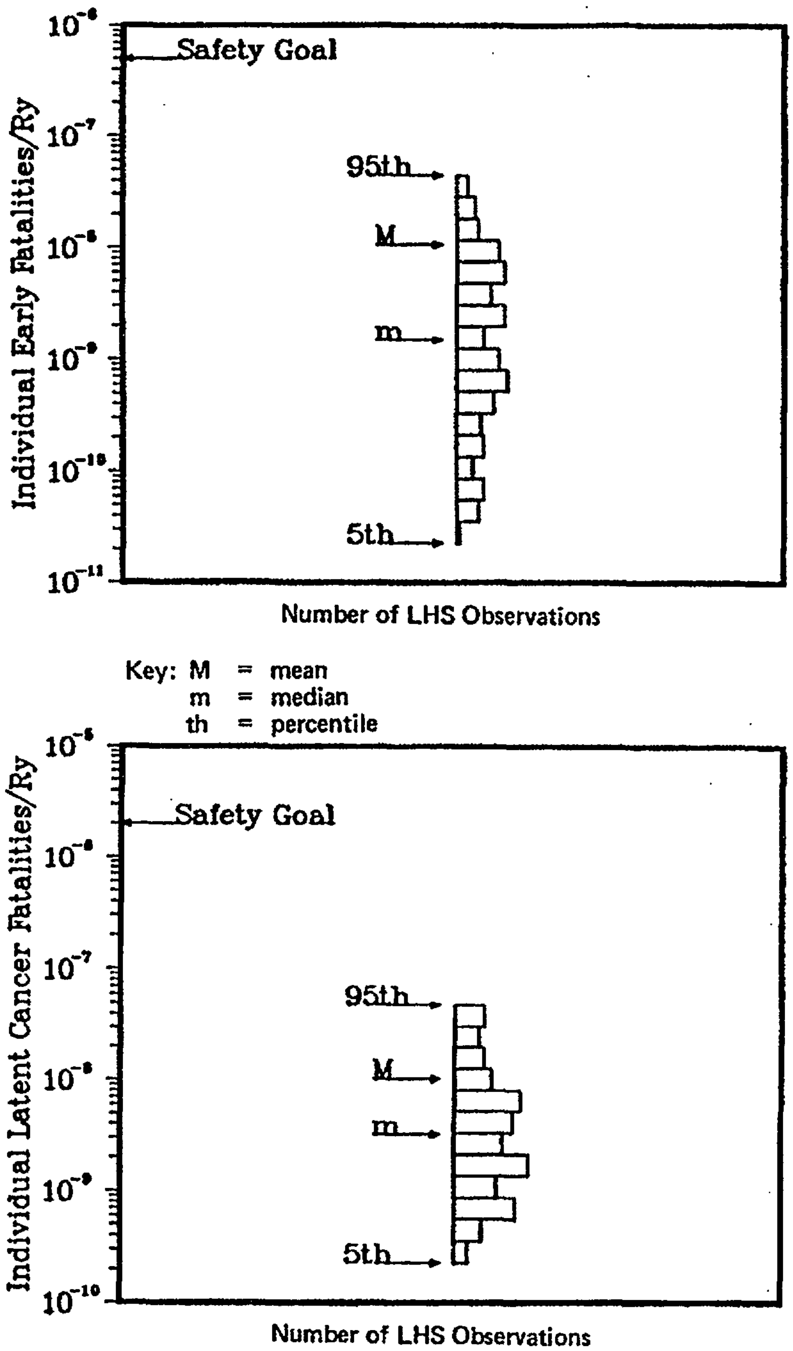
Note: As discussed in Reference 5.3, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 5.9 Early and latent cancer fatality risks at Sequoyah (internal initiators).



Note: As discussed in Reference 5.3, 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 5.10 Population dose risks at Sequoyah (internal initiators).



Note: As discussed in Reference 5.3, 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 5.11 Individual early and latent cancer fatality risks at Sequoyah (internal initiators).

early and latent cancer fatality risks, while quite low in absolute value, are higher than those from the Surry plant analysis (see Chapter 3). Other risk measure estimates are slightly higher than the Surry estimates. The individual early fatality and latent cancer fatality risks are well below the NRC safety goals. Detailed comparisons of results are provided in Chapter 12.

The risk results shown in Figure 5.9 have been analyzed to identify the relative contributions to mean risk of plant damage states and accident progression bins. These results are presented in Figures 5.12 and 5.13. As may be seen, the dominant contributor of early fatality risk is the bypass accident group, and particularly the interfacing-system LOCA (the V sequence); whereas the largest contributions to the latent cancer fatality risk came from the station blackout and bypass accident groups. For early fatality risk, the dominant contributor to risk is from accident sequences where the containment is bypassed, whereas, for latent cancer fatality risk, major accident progression bin contributors are bypass accidents and early containment failures. The accident progression bin involving accidents with no vessel breach appears as a contributor to early and latent cancer fatality risks. This bin possesses risk potential because of early containment failure due to hydrogen events from loss of offsite power in which ac power is recovered and breach is arrested and also from accidents involving steam generator tube rupture in which vessel breach is arrested.

5.6.2 Important Plant Characteristics (Risk)

Sequoyah risk analysis indicates that bypass sequences dominate early fatality risk. Timing is a key factor in this sequence in relation to evacuation. The release characteristics also contribute to the large effect of early fatalities because of the large magnitude of unmitigated source terms and the low energy of the first release. The low energy plume is not lofted over the evacuees but is held low to the ground after release. Another class of accidents that is important to early fatality risk is station blackout. It is the early containment failure (that is, failure of containment at and before vessel breach) associated with this accident class that contributes to early fatality risk.

An interfacing-system LOCA at Sequoyah will discharge into the auxiliary building where decontamination by automatically activated fire sprays is likely. Neither the probability of actuation nor the decontamination factor has been well established. The effects of an interfacing-system LOCA could either be higher or lower than those that have been calculated in this study.

Approximately equal contributions to latent cancer fatality risk come from station blackout and bypass. The bypass sequences contribute because of the large source terms and the bypass of any mitigating systems. The only other major contribution to latent cancer fatality comes from the LOCA sequences, mainly due to containment failures at vessel breach with high (> 200 psia) reactor coolant system pressure.

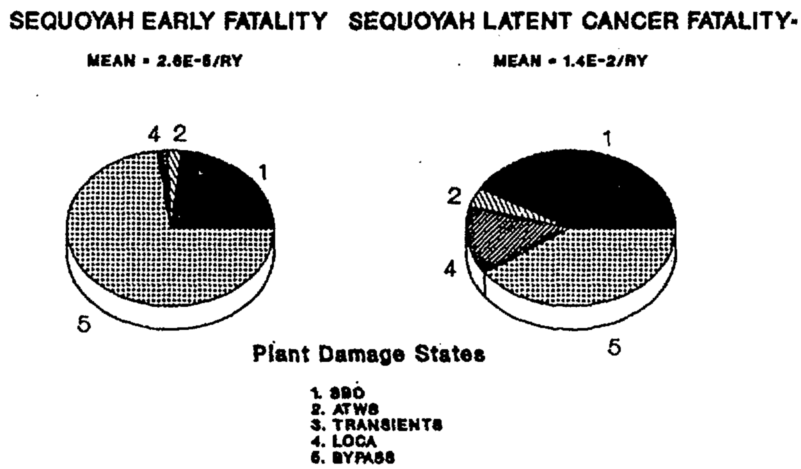


Figure 5.12 Major contributors (plant damage states) to mean early and latent cancer fatality risks at Sequoyah (internal initiators).

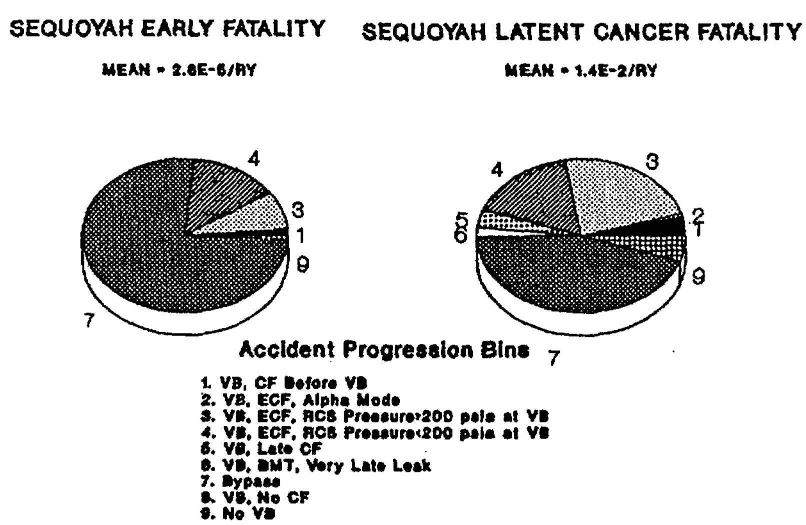


Figure 5.13 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Sequoyah (internal initiators).

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5.5 Tennessee Department of Transportation, "Evacuation Time Estimates with the Plume Exposure Pathway Emergency Planning Zone," prepared for Sequoyah Nuclear Plant, June 1987.

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6. GRAND GULF PLANT RESULTS

6.1 Summary Design Information

The Grand Gulf Nuclear Station is a General Electric boiling water reactor (BWR-6) unit of 1250 MWe capacity housed in a Mark III containment. Grand Gulf Unit 1, constructed by Bechtel Corporation, began commercial operation in July 1985 and is operated by Entergy Operations. Some important design features of the Grand Gulf plant are described in Table 6.1. A general plant schematic is provided in Figure 6.1.

This chapter provides a summary of the results obtained in the detailed risk analyses underlying this report (Refs. 6.1 and 6.2). A discussion of perspectives with respect to these results is provided in Chapters 8 through 12.

6.2 Core Damage Frequency Estimates

6.2.1 Summary of Core Damage Frequency Estimates

The core damage frequency and risk analyses performed for this study considered accidents initiated only by internal events (Ref. 6.1). The core damage frequency results obtained are provided in tabular form in Table 6.2 and in graphical form, displayed as a histogram, in Figure 6.2. (Section 2.2.2 discusses histogram development.) This study calculated a total median core damage frequency from internal events of 1.2E-6 per year.

The Grand Gulf plant was previously analyzed in the Reactor Safety Study Methodology Applications Program (RSSMAP) (Ref. 6.3). A point estimate core damage frequency of 3.6E-5 from internal events was calculated in that study. A point estimate core damage frequency of 2.1E-6 was calculated in this analysis for purposes of comparison. A point estimate is calculated from the sum of all the cut-set frequencies, where each of the cut-set frequencies is the product of the point estimates (usually means) of the events in the cut sets.

6.2.1.1 Internally Initiated Accident Sequences

A detailed description of accident sequences important at the Grand Gulf plant is provided in Reference 6.1. For this report, the accident sequences described in that reference have been divided into two summary plant damage states. These are:

- Station blackout, and
- Anticipated transients without scram (ATWS).

The relative contributions of these groups to mean internal-event core damage frequency at Grand Gulf are shown in Figure 6.3. It may be seen that station blackout accident sequences as a class are the largest contributors to core damage frequency. It should be noted that the plant configuration as analyzed does not reflect the implementation of the station blackout rule.

Within the general class of station blackout accidents, the more probable combinations of failures leading to core damage are:

- Loss of offsite power occurs followed by the successful cycling of the safety relief valves (SRVs). Onsite ac power fails because all three diesel generators fail to start and run as a result of either hardware or common-cause faults. The loss of all ac power (i.e., station blackout) results in the loss of all core cooling systems (except for the reactor core isolation cooling (RCIC) system) and all containment heat removal systems. The RCIC system, which is ac independent, independently fails to start and run. All core cooling is lost, and core damage occurs in approximately 1 hour after offsite power is lost.
- Station blackout accident that is similar to the one described above except that one SRV fails to reclose and sticks open. Core damage occurs in approximately 1 hour after offsite power is lost.

In addition to these two short-term accident scenarios, this study also considered long-term station blackout accidents. In these accidents, loss of offsite power occurs and all three diesel generators fail to start or run. The safety relief valves cycle successfully and RCIC starts and maintains proper coolant level within the reactor vessel. However, ac power is not restored in these long-term scenarios, and RCIC eventually fails because of high turbine exhaust pressure, battery depletion, or other long-term effects. Core damage occurs approximately 12 hours after offsite power is lost.

Table 6.1 Summary of design features: Grand Gulf Unit 1.

1. Coolant Injection Systems	a.	High-pressure core spray (HPCS) system provides coolant to reactor vessel during accidents in which system pressure remains high or low, with 1 train and 1 MDP.*
	b.	Reactor core isolation cooling system provides coolant to the reactor vessel during accidents in which system pressure remains high, with 1 train and 1 TDP.*
	c.	Low-pressure core spray system provides coolant to the reactor vessel during accidents in which vessel pressure is low, with 1 train and 1 MDP.*
	d.	Low-pressure coolant injection system provides coolant to the reactor vessel during accidents in which vessel pressure is low, with 3 trains and 3 pumps.
	e.	Standby service water crosstie system provides coolant makeup source to the reactor vessel during accidents in which normal sources of emergency injection have failed, with 1 train and 1 pump (for crosstie).
	f.	Firewater system is used as a last resort source of low-pressure coolant injection to the reactor vessel, with 3 trains, 1 MDP,* 2 diesel-driven pumps.
	g.	Control rod drive system provides backup source of high-pressure injection, with 2 pumps/238 gpm (total)/1103 psia.
	h.	Automatic depressurization system (ADS) depressurizes the reactor vessel to a pressure at which the low-pressure injection systems can inject coolant to the reactor vessel, with 8 relief valves/capacity of 900,000 lb/hr. In addition, there are 12 non-ADS relief valves.
	i.	Condensate system used as a backup injection source.
2. Heat Removal Systems	a.	Residual heat removal/suppression pool cooling system removes decay heat from the suppression pool during accidents, with 2 trains and 2 pumps.
	b.	Residual heat removal/shutdown cooling system removes decay heat during accidents in which reactor vessel integrity is maintained and reactor is at low pressure, with 2 trains and 2 pumps.
	c.	Residual heat removal/containment spray system suppresses pressure in the containment during accidents, with 2 trains and 2 pumps.
3. Reactivity Control Systems	a.	Control rods.
	b.	Standby liquid control system, with 2 parallel positive displacement pumps rated at 43 gpm per pump.

*TDP -Turbine-Driven Pump
MDP - Motor-Driven Pump

Table 6.1 (Continued)

4. Key Support Systems	a.	dc power with 12-hour station batteries.
	b.	Emergency ac power, with 2 diesel generators and third diesel generator dedicated to HPCS but with crossties.
	c.	Suppression pool makeup system provides water from the upper containment pool to the suppression pool following a LOCA.
	d.	Standby service water provides cooling water to safety systems and components.
5. Containment Structure	a.	BWR Mark III.
	b.	1.67 million cubic feet.
	c.	15 psig design pressure.
6. Containment Systems	a.	Containment venting is used when suppression pool cooling and containment sprays have failed to reduce primary containment pressure.
	b.	Hydrogen igniter system prevents the buildup of large quantities of hydrogen inside the containment during accident conditions.

Within the general class of ATWS accidents, the most probable combination of failures leading to core damage is:

- Transient initiating event occurs followed by a failure to trip the reactor because of mechanical faults in the reactor protection system (RPS). The standby liquid control system (SLCS) is not actuated and the high-pressure core spray (HPCS) system fails to start and run because of random hardware faults. The reactor is not depressurized and therefore the low-pressure core cooling system cannot inject. All core cooling is lost; core damage occurs in approximately 20 to 30 minutes after the transient initiating event occurs.

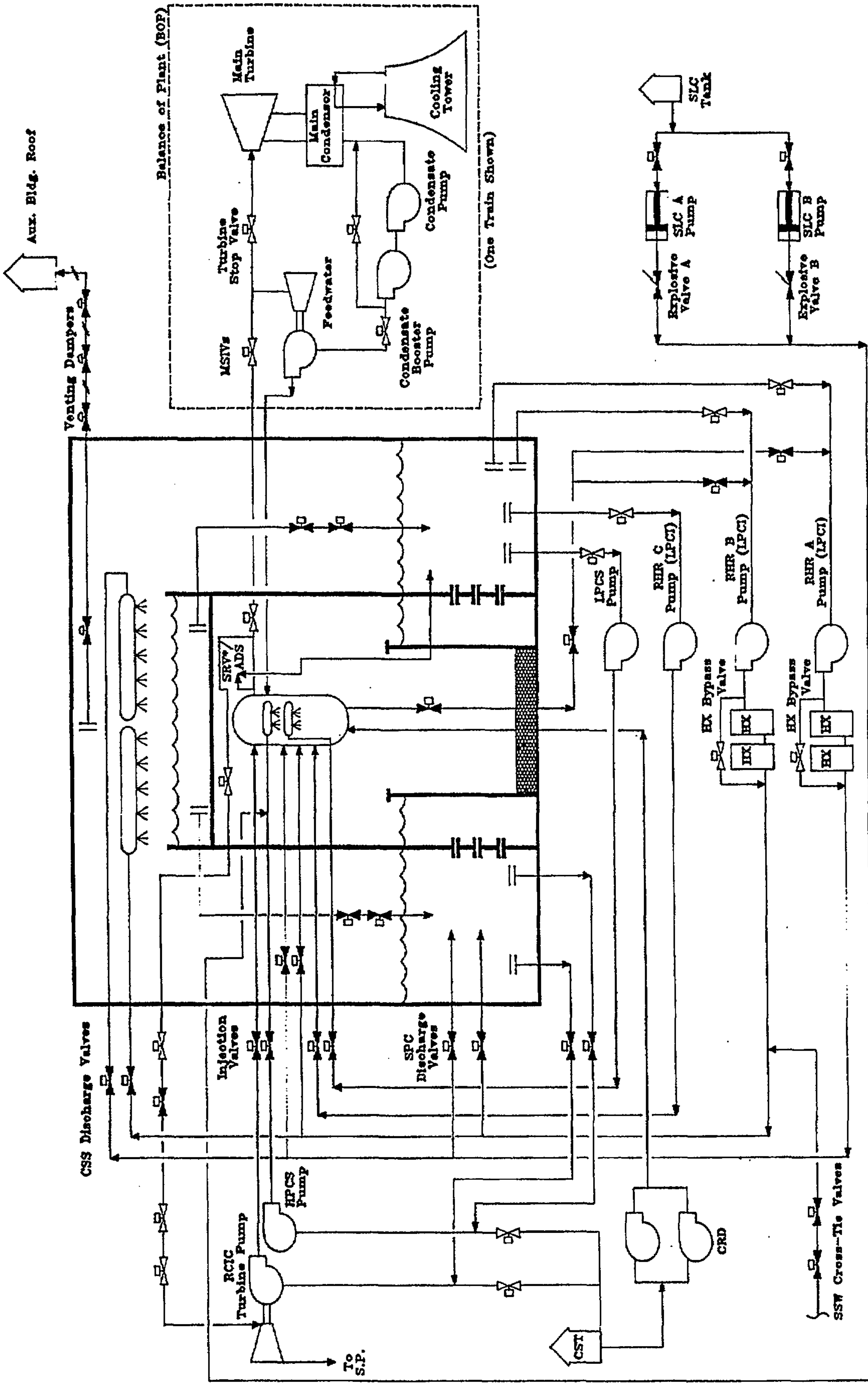
6.2.2 Important Plant Characteristics (Core Damage Frequency)

Characteristics of the Grand Gulf plant design and operation that have been found to be important in the analysis of core damage frequency include:

1. **Firewater System as Source of Coolant Makeup**

   The firewater system as a core coolant injection system can be used as a backup (last resort) source of low-pressure coolant injection to the reactor vessel. The system has two diesel-driven pumps, making it operational under station blackout conditions as long as dc power is available. The potential use of this system is estimated to reduce the total core damage frequency by approximately a factor of 1.5.

   The reason for the relatively small impact on the total core damage frequency is twofold. The firewater system is a low-pressure system; the reactor pressure must be maintained below approximately 125 psia for firewater to be able to inject. If an accident occurs in which core cooling is immediately lost, the core becomes uncovered in less time than that required to align and activate the firewater system. If core cooling is provided and then lost in the long term (e.g., at approximately greater than 4 hours after the start of the accident), firewater can provide sufficient makeup to prevent core damage. However, the dominant sequences at Grand Gulf are accidents where core cooling is lost immediately.



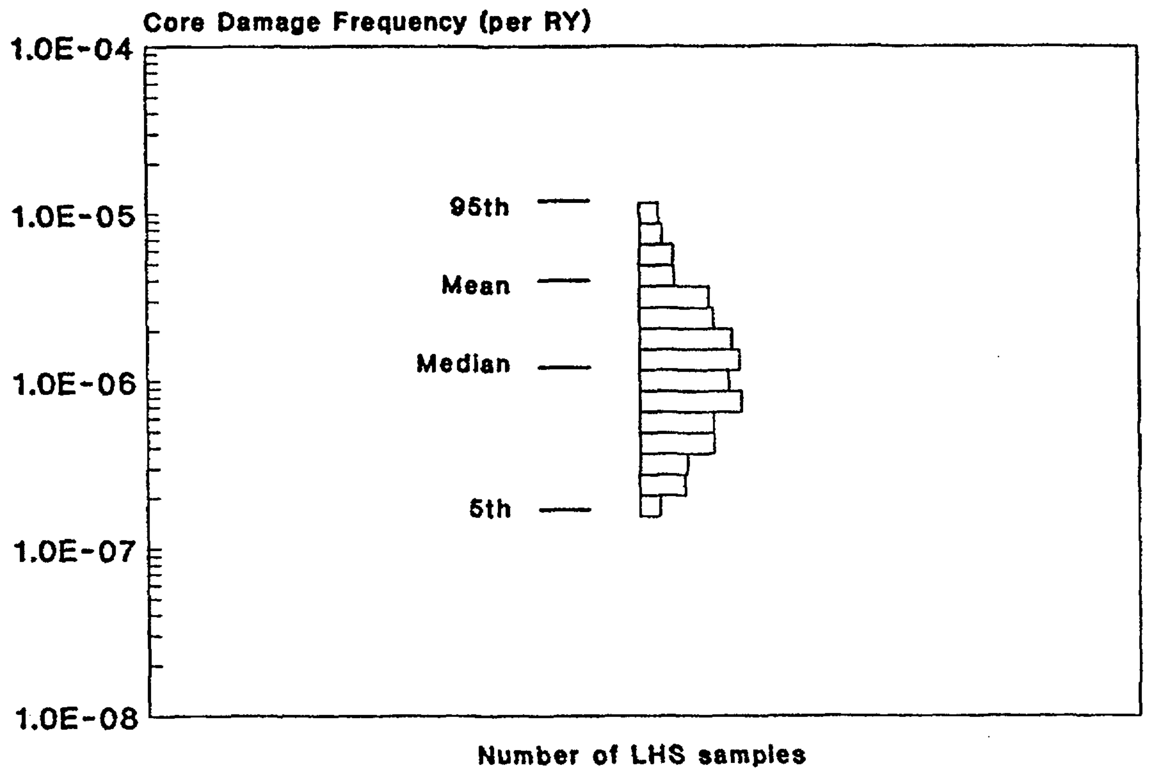
*Typical arrangement (8 ADS SRVs and 12 non-ADS SRVs)

Figure 6.1 Grand Gulf plant schematic.

Table 6.2 Summary of core damage frequency results: Grand Gulf.*

	5%	Median	Mean	95%
Internal Events	1.7E-7	1.2E-6	4.0E-6	1.2E-5
ATWS	8.5E-10	1.9E-8	1.1E-7	5.1E-7
Station Blackout	1.3E-7	1.1E-6	3.9E-6	1.1E-5

*As discussed in Reference 6.4, 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).



Note: As discussed in Reference 6.4, 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 6.2 Internal core damage frequency results at Grand Gulf.

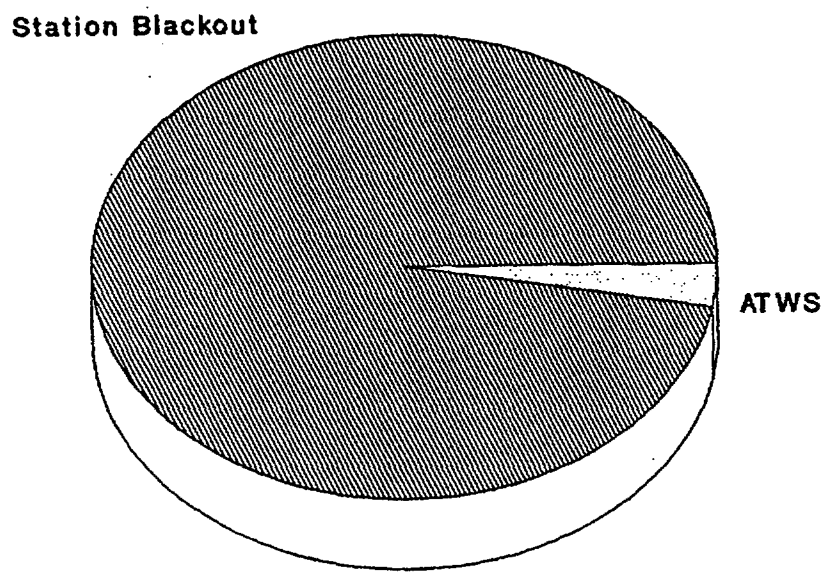


Figure 6.3 Contributors to mean core damage frequency from internal events at Grand Gulf.

2. **High-Pressure Core Spray (HPCS) System**

The HPCS system consists of a single train with motor-operated valves and a motor-driven pump and provides coolant to the reactor vessel during accidents in which pressure is either high or low. The bearings and seals of the HPCS pump are cooled by the pumped fluid. If the temperature of this water exceeds design limits, the potential exists for the HPCS pump to fail. The bearings are designed to operate for no more than 24 hours at a temperature of 350°F. The peak temperature achieved in any of the accidents analyzed is approximately 325°F. Even if the seals were to experience some leakage, the resultant HPCS room environment would not adversely affect the operability of the pump. The availability of an HPCS system with such design characteristics is estimated to reduce the core damage frequency by approximately a factor of 7. The HPCS is powered by a dedicated diesel generator when required so that this system is truly an independent system.

3. **Capability of Pumps to Operate with Saturated Water**

The emergency core cooling pumps that depend on the pressure-suppression pool as their water source during accident conditions have been designed to pump saturated water. Thus, if the pool becomes saturated because of containment venting or containment failure, the core cooling systems are not lost but can continue to cool the reactor core.

4. **Redundancy and Diversity of Water Supply Systems**

At Grand Gulf, there are many redundant and diverse systems to provide water to the reactor vessel. They include:

HPCS with 1 pump;

Reactor core isolation cooling (RCIC) with 1 pump;

Control rod drive (CRD) with 2 pumps (both are required for core cooling);

Condensate with 3 pumps;

Low-pressure core spray (LPCS) with 1 pump;

Low-pressure coolant injection (LPCI) with 3 pumps;

Standby service water (SSW) crosstie with 1 pump; and

Firewater system with 3 pumps.

Because of the redundancy of systems for LOCAs and transients, core cooling loss as a result of independent random failures is of low probability. However, in a station blackout, except for RCIC and firewater, the core cooling systems are lost with a probability of unity because they require ac power.

5. **Redundancy and Diversity of Heat Removal Systems**

At Grand Gulf there are several diverse means for heat removal. These systems are:

Main steam/feedwater system with 3 trains;

Suppression pool cooling mode of residual heat removal (RHR) with 2 trains;

Shutdown cooling mode of RHR with 2 trains;

Containment spray system mode of RHR with 2 trains; and

Containment venting with 1 train.

Although the various modes of RHR have common equipment (e.g., pumps), there is still enough redundancy and diversity that, for non-station-blackout accidents, independent random failures again are small contributors to the core damage frequency.

6. **Automatic and Manual Depressurization System**

The automatic depressurization system (ADS) is designed to depressurize the reactor vessel to a pressure at which the low-pressure injection systems can inject coolant to the reactor vessel. The ADS consists of eight safety relief valves capable of being manually opened. The operator may manually initiate the ADS or may depressurize the reactor vessel, using the 12 relief valves that are not connected to the ADS logic. The ADS valves are located inside the containment.

6.2.3 Important Operator Actions

The emergency operating procedures (EOPs) at Grand Gulf direct the operator to perform certain actions depending on the plant conditions or symptoms (e.g., reactor vessel level below the top of active fuel). Different accident sequences can have similar symptoms and therefore the same "recovery" actions. Operator actions that are important include the following:

- Actuate core cooling

  In an accident where feedwater is lost (which includes condensate), the reactor water level starts to decrease. When Level 2 (-41.6 inches) is reached, high-pressure core spray (HPCS) and reactor core isolation cooling (RCIC) should be automatically actuated. If Level 1 (-150.3 inches) is reached, the ADS should occur with automatic actuation of the low-pressure core spray (LPCS) and low-pressure coolant injection (LPCI). If the reactor level sensors are miscalibrated, these systems will not automatically actuate. The operator has many other indications to determine both the reactor water level and the fact that core coolant makeup is not occurring. Manual actuation of these systems is required if such failures occur in order to prevent core damage.

- Establish containment heat removal

  Besides core cooling, the operator must also establish containment heat removal (CHR). If an accident occurs, the EOPs direct the operator to initiate the suppression pool cooling mode of RHR when the suppression temperature reaches 95°F. The operator closes the LPCI valves and the heat exchanger bypass valves and opens the suppression pool discharge valves. He also ensures that the proper service water system train is operating. With suppression pool cooling (SPC) functioning, CHR is being performed. If system faults preclude the use of SPC, the operator has other means to provide CHR. He can actuate other modes of RHR such as shutdown cooling or containment spray, or the operator can vent the containment to remove the energy.

- Establish room cooling through natural circulation

  The heating, ventilating, and air conditioning (HVAC) system provides room cooling support to a variety of systems. If HVAC is lost, design limits can be exceeded and equipment (i.e., pumps) can fail. If these conditions occur, the operator can open doors to certain rooms and establish a natural circulation/ventilation that prevents the room temperature from exceeding the design limits of the equipment.

For station blackout accidents, there are certain actions that can be performed by the operating crew as follows:

- Crosstie division 1 or 2 loads to HPCS diesel generator

  In a station blackout where the HPCS diesel generator is available, the operator can choose to crosstie this diesel to one of the other divisions. The operator might choose this option when (1) the HPCS system fails and core cooling is required, or (2) in the long term (e.g., longer than 8 hours) containment heat removal is required to prevent containment failure. If the operator chooses to crosstie, the operator must shed all the loads from the HPCS diesel and then open and close certain breakers. He can then load certain systems from either division 1 or from division 2.

- Align firewater

  In an accident, particularly station blackout, where core cooling was initially available (for approximately 4 hours) and then lost, the firewater system can provide adequate core cooling. The operator must align the firewater hoses to the proper injection lines (described in the procedure) and then open the injection valves.

- Depressurize reactor via RCIC steam line

  In a station blackout, the diesel generators have failed and only dc power is available (in certain sequences). If core cooling is being provided with firewater, then the reactor must remain at low pressure, which requires that at least one safety relief valve (SRV) must remain open. For the SRV to remain open, dc power is required. However, without the diesel generator recharging the battery, the battery will eventually deplete, the SRV will close, and the reactor will repressurize, which causes the loss of the firewater. The operator can maintain the reactor pressure low by opening the valves on the RCIC steam line. This provides a vent path from the reactor to the suppression pool.

- Recovering ac power

  Station blackout is caused by the loss of all ac power, both offsite and onsite power. Restoring offsite power or repairing the diesel generators was included in the analysis. The quantification of these human failure events was derived from historical data (i.e., actual time required to perform these repairs) and not by performing human reliability analysis on these events.

Transients where reactor trip does not occur (i.e., ATWS) involve accident sequences where the phenomena are more complex. The operator actions were evaluated in more detail (Ref. 6.5) than for the regular transient-initiated accident. These actions include the following:

- Manual scram

  A transient occurs that demands the reactor to be tripped, but the reactor protection system (RPS) fails because of electrical faults. The operator can then manually trip the reactor by first rotating the collar on proper scram buttons and then depressing the buttons, or he can put the reactor mode switch in the "shutdown" position.

- Insert rods manually

  If the electrical faults fail both the RPS and the manual trip, the operator can manually insert the control rods one a time.

- Actuate standby liquid control (SLC) system

  With the reactor not tripped, reactor power remains high; the reactor core is not at decay heat levels. This can present problems since the containment heat removal systems are only designed to decay heat removal capacity. However, the SLC system (manually actuated) injects sodium pentaborate that reduces reactor power to decay heat levels. The EOPs direct the operator to actuate SLC if the reactor power is above 4 percent and before the suppression pool temperature reaches 110°F. The operator obtains the SLC keys (one per pump) from the shift supervisor's desk, inserts the keys into the switches, and turns both to the "on" position.

- Inhibit automatic depressurization system (ADS)

  In an ATWS condition, the operator is directed to inhibit the ADS if he has actuated

SLC. The operator must put both ADS switches (key locked) in the inhibit mode.

- Manually depressurize reactor

  If HPCS fails, inadequate high-pressure core cooling occurs. When Level 1 is reached, ADS will not occur because the ADS was inhibited, and the operator must manually depressurize so that low-pressure core cooling can inject. The operator can either press the ADS button (which overrides the inhibit) or manually open one SRV at a time.

6.2.4 Important Individual Events and Uncertainties (Core Damage Frequency)

As discussed in Chapter 2, the process of developing a probabilistic model of a nuclear power plant involves the combination of many individual events (initiators, hardware failures, operator errors, etc.) into accident sequences and eventually into an estimate of the total frequency of core damage. After development, such a model can also be used to assess the importance of the individual events. The detailed studies underlying this report have been analyzed using several event importance measures. The results of the analyses using two measures, "risk reduction" and "uncertainty" importance, are summarized below.

- Risk (core damage frequency) reduction importance measure (internal events)

  The risk-reduction importance measure is used to assess the change in core damage frequency as a result of setting the probability of an individual event to zero. Using this measure, the following individual events were found to cause the greatest reduction in core damage frequency if their probabilities were set to zero.

  - Loss of offsite power initiating event. The core damage frequency would be reduced by approximately 92 percent.

  - Failure to restore offsite power in 1 hour. The core damage frequency would be reduced by approximately 70 percent.

  - Failure of the RCIC turbine-driven pump to run. The core damage frequency would be reduced by approximately 48 percent.

  - Failure to repair hardware faults of diesel generator in 1 hour. The core damage frequency would be reduced by approximately 46 percent.

  - Failure of a diesel generator to start. The core damage frequency would be reduced by approximately 23 to 32 percent, depending on the diesel generator.

  - Common-cause failure of the vital batteries. The core damage frequency would be reduced by approximately 20 percent.

- Uncertainty importance measure (internal events)

  A second importance measure used to evaluate the core damage frequency analysis results is the uncertainty importance measure. For this measure, the relative contribution of the uncertainty of individual events to the uncertainty in total core damage frequency is calculated. Using this measure, the following events were found to be most important:

  - Loss of offsite power;

  - Failure of the diesel generators to run, given start;

  - Individual and common-cause failure of the diesel generators to start;

  - Standby service water motor-operated valves (MOVs) fail to open; and

  - High-pressure core spray and RCIC MOVs fail to function.

6.3 Containment Performance Analysis

6.3.1 Results of Containment Performance Analysis

The Grand Gulf pressure-suppression containment design is of the Mark III type in which the reactor vessel, reactor coolant circulating loops, and other branch connections to the reactor coolant system are housed within the drywell structure. The drywell structure in turn is completely contained within an outer containment structure with the two volumes communicating through the water-filled vapor suppression pool. The outer containment building is a steel-lined reinforced concrete structure with a volume of 1.67 million cubic feet that is designed for a peak pressure of 15 psig resulting from a reactor coolant system loss-of-coolant accident. For this same design basis accident, the inner concrete drywell structure is designed for a peak pressure of 30 psig. The mean failure pressure for Grand Gulf's containment structure has been estimated to be 55 psig. This estimated containment failure pressure for Grand Gulf is much lower than the Peach Bottom Mark I estimated failure pressure of 148 psig; however, Grand Gulf's free volume is several times larger. The availability of Grand Gulf's large volume removed the design basis need to inert the containment against failure from hydrogen combustion following design basis accidents; however, subsequent severe accident considerations after the TMI accident resulted in the installation of hydrogen igniters. For the severe accident sequences developed in this analysis, hydrogen combustion remains the major threat to Grand Gulf's containment integrity (in the station blackout accidents dominating the frequency of core damage, igniters are not operable). Similar to other containment design analyses, the estimate of where and when Grand Gulf's containment system will fail relied heavily on the use of expert judgment to interpret the limited data available.

The potential for early containment and/or drywell failure for Grand Gulf as compared to Peach Bottom's Mark I suppression-type containment involves significantly different considerations. Of particular significance with regard to the potential for large radioactive releases from Grand Gulf is the prediction of the combined probabilities of simultaneous early containment and drywell failures, which in turn produce a direct radioactive release path to the environment. The results of these analyses for Grand Gulf are shown in Figures 6.4 and 6.5. Figure 6.4 displays information in which the eight conditional probabilities of containment-related accident progression bins; e.g., VB-early CF-no SPB, are presented for each of four plant damage states, e.g., ATWS. This information indicates that, on a plant damage state frequency-weighted average* for internally initiated events, there are mean conditional probabilities of (1) 0.23 that the integrity of the drywell and the outer containment will be sufficiently affected that substantial bypass of the suppression pool will occur; (2) 0.24 for early containment failure with no bypass of the suppression pool pathway from the drywell; (3) 0.12 for late containment failure with pool bypass; (4) 0.23 for late containment failure but no pool bypass; and (5) 0.09 for no containment failure.

*Each value in the column in Figure 6.4 labeled "All" is a frequency-weighted average obtained by summing the products of individual accident progression bin conditional probabilities for each plant damage state and the ratio of the frequency of that plant damage state to the total core damage frequency.

Further examination of these data, broken down on the basis of the timing of reactor vessel breach and the nature of the containment threat, indicate: (1) prior to reactor vessel breach, hydrogen combustion and slow steam overpressurization effects lead to frequency-weighted mean conditional probabilities of containment failure of 0.20 and 0.05, respectively; (2) at reactor vessel breach, hydrogen combustion effects lead to a 0.24 conditional mean probability of containment failure; (3) prior to reactor vessel breach, hydrogen combustion effects lead to 0.12 conditional mean probability of drywell failure; (4) at reactor vessel breach, steam explosion and direct containment heating effects can lead to pedestal failures and a 0.16 conditional mean probability of drywell failure from both pedestal and overpressure effects; and (5) dynamic loads from hydrogen detonations have a small effect on the structural integrity of either the containment or the drywell.

Figure 6.5 further displays plots of Grand Gulf's conditional probability distribution for each plant damage state, thereby providing the estimated range of uncertainties in the outer containment failure predictions. The important conclusions that can be drawn from the information are (1) there is a relatively high mean conditional probability of early containment failure with a large bypass of the suppression pool's scrubbing effects, i.e., 0.23; (2) there is a high mean probability of early containment failure, i.e., 0.48; and (3) the principal threat to the combined efficacy of the Mark III containment and drywell is hydrogen combustion effects.

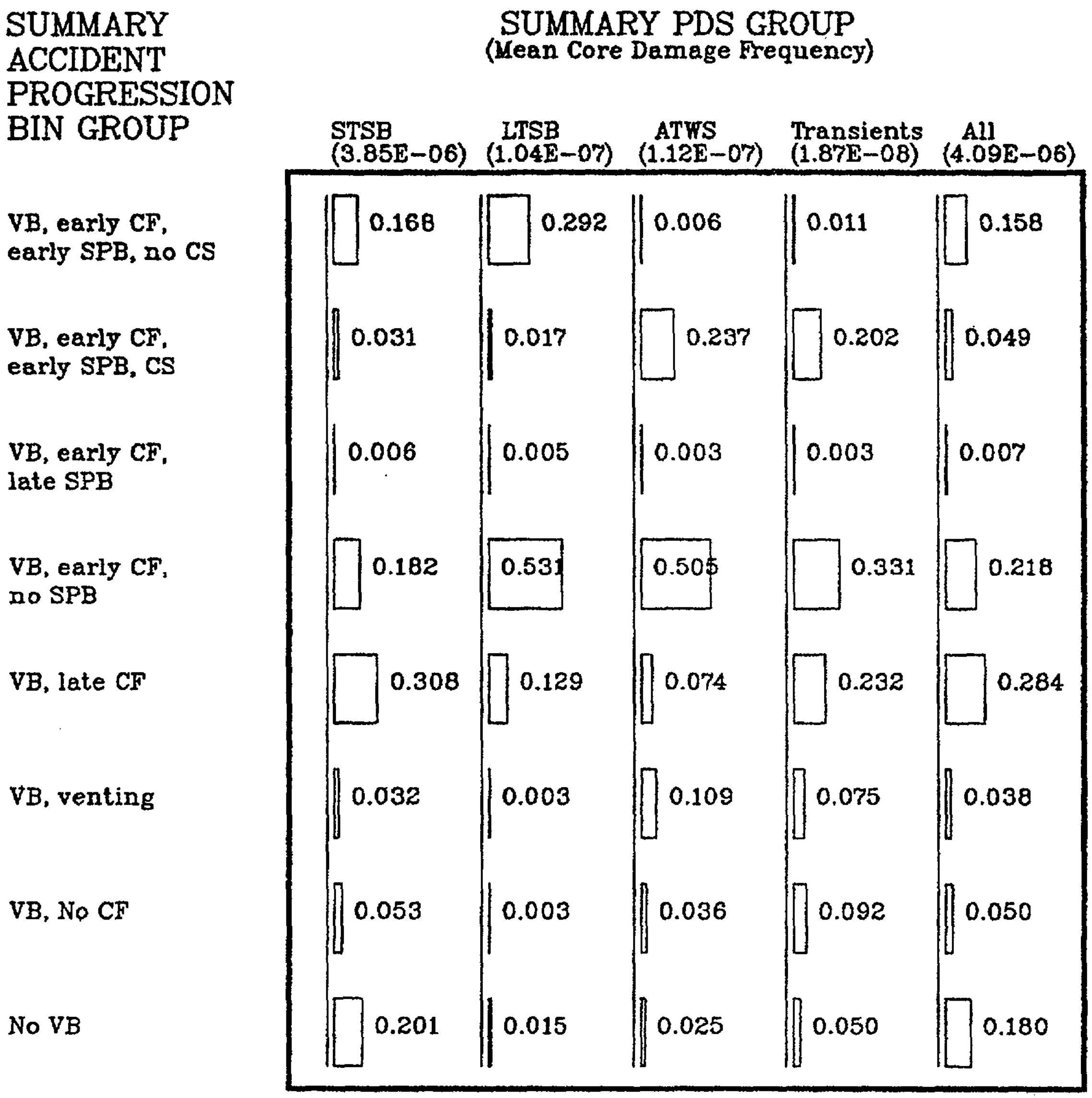
Additional discussions on containment performance (for all studied plants) are provided in Chapter 9.

6.3.2 Important Plant Characteristics (Containment Performance)

Characteristics of the Grand Gulf design and operation that are important during core damage accidents include:

1. **Drywell-Wetwell Configuration**

   With the reactor vessel located inside the drywell, which in turn is completely surrounded by the outer containment building, there needs to be a combination of failures in both structures to provide a direct release path to the environment that bypasses the suppression pool, e.g., hydrogen combustion



CF = Containment Failure
CS = Containment Sprays
CV = Containment Venting
SPB = Suppression Pool Bypass
VB = Vessel Breach

Figure 6.4 Conditional probability of accident progression bins at Grand Gulf.

impairing the function of both the drywell and containment.

2. **Containment Volume**

The Grand Gulf containment volume is much larger than that of a Mark I containment and as such can accommodate significant quantities of noncombustible gases before failure even though its estimated failure pressure is less than half that of a Mark I containment. Its low design pressure, however, makes it susceptible to failure from hydrogen combustion effects in those cases where the igniters are not working.

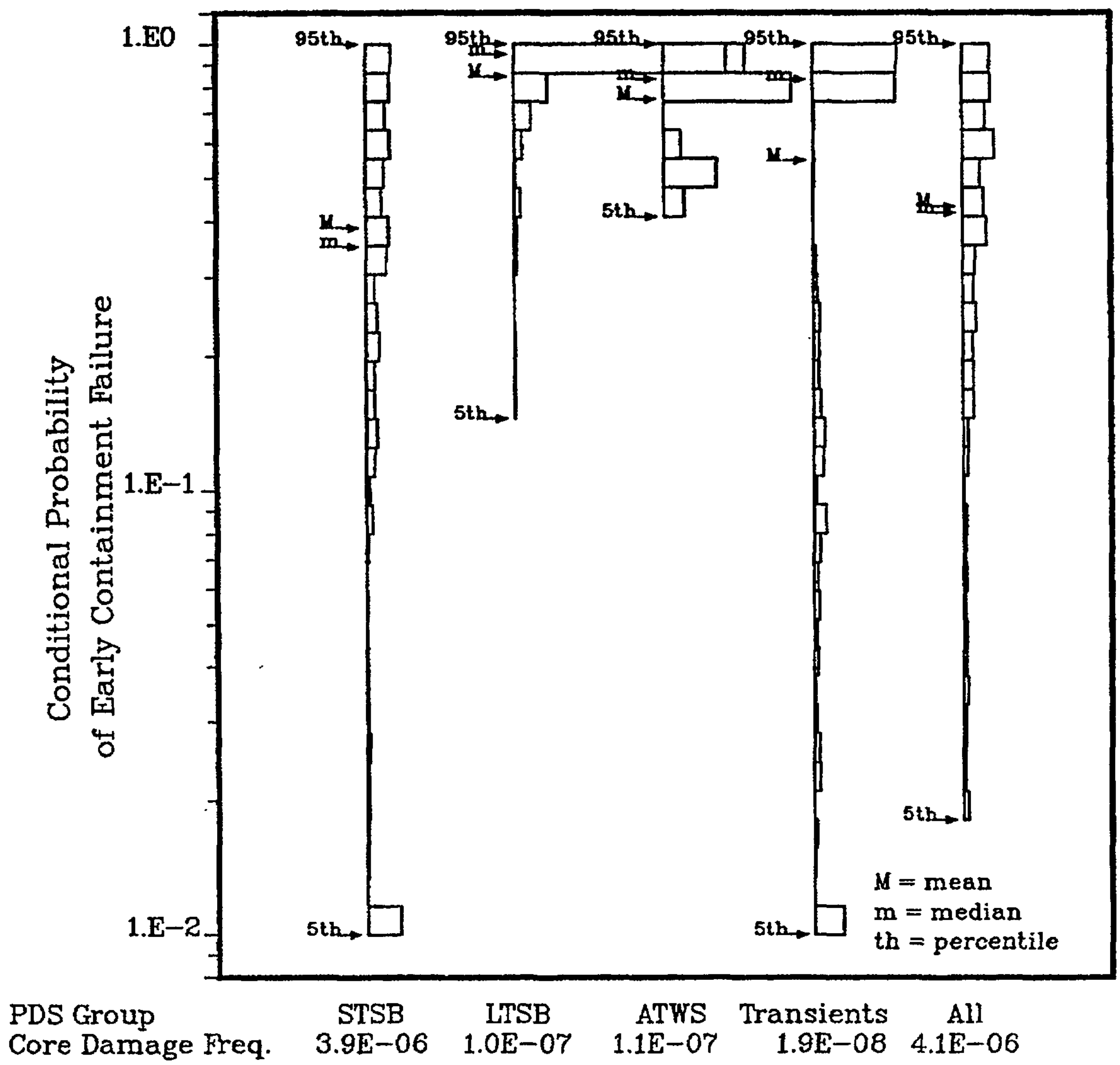


Figure 6.5 Conditional probability distributions for early containment failure at Grand Gulf.

3. **Hydrogen Ignition System**

The Grand Gulf containment hydrogen ignition system is capable of maintaining the concentration of hydrogen from severe accidents in manageable proportions for many severe accidents. However, for station blackout accident sequences, the igniter system is not operable. When power is restored, the ignition system will be initiated; potentially the containment has high hydrogen concentrations. Some potential then exists for a deflagration causing simultaneous failures of both the containment building and the drywell structure.

4. **Containment Spray System**

The Grand Gulf containment spray system has the capability to condense steam and reduce the amount of radioactive material released to the environment for specific accident sequences. However, for some sequences, i.e., loss of ac power, its eventual initiation upon power recovery and that of the hydrogen ignition system could result in subsequent hydrogen combustion that has some potential to fail the containment and drywell.

6.4 Source Term Analysis

6.4.1 Results of Source Term Analysis

A key difference between the Peach Bottom (Mark I) design and Grand Gulf (Mark III) design is the wetwell/drywell configuration. If the drywell remains intact in the accident and the mode of containment failure does not result in loss of the suppression pool, leakage to the environment must pass through the pool and be subject to decontamination.

Figures 6.6 and 6.7 illustrate the effect of drywell integrity in mitigating the environmental release of radionuclides for early containment failure. In Figure 6.6, both the drywell and the containment fail early and sprays are not available. The median release for the volatile radionuclides is approximately 10 percent, indicative of a large release with the potential for causing early fatalities. For the early containment failure accident progression bin with the drywell intact, as illustrated in Figure 6.7, the environmental source terms are reduced, since the flow of gases escaping the containment after vessel breach must also pass through the suppression pool before being released to the environment.

Additional discussion on source term perspectives (for all studied plants) is provided in Chapter 10.

6.4.2 Important Plant Characteristics (Source Term)

1. **Suppression Pool**

   The pressure-suppression pool at Grand Gulf provides the potential for substantial mitigation of the source terms in severe accidents. Since transient-initiated accidents represent a large contribution to core damage frequency, the in-vessel release of radionuclides is almost always subject to pool decontamination. Only a fraction of such accident sequences (in which a vacuum breaker sticks open in a safety relief valve discharge line) releases radionuclides directly to the drywell in this phase of the accident. The pool decontamination factors used for the Grand Gulf design for the in-vessel release range from 1.1 to 4000, with a median of 60. For the ex-vessel release component, the pool is less effective. The decontamination factors range from 1 to 90 with a median of 7.

2. **Wetwell-Drywell Configuration**

   If the drywell remains intact in a severe accident at Grand Gulf, the radionuclide release would be forced to pass through the suppression pool and the source term would be substantially mitigated. However, the likelihood of drywell failure is estimated to be quite significant, such that early failure with suppression pool bypass occurs approximately one-quarter of the time if core melting and vessel breach occur.

3. **Pedestal Flooding**

   The pedestal region communicates with the drywell region through drains in the drywell floor. The amount of water in the pedestal region depends on whether the upper water pool has been dumped into the suppression pool, on the quantity of condensate storage that has been injected into the containment, and on the transient pressurization of the containment building resulting from hydrogen burns. The effect of water in the pedestal is either to result in debris coolability or to mitigate the source term to containment of the radionuclides released during core-concrete interaction. Water in the pedestal does, however, also introduce some potential for a steam explosion that can damage the drywell.

4. **Containment Sprays**

   Containment sprays can have a mitigating effect on the release of radionuclides under conditions in which both the containment and drywell have failed. In other accident scenarios in which the in-vessel and ex-vessel releases must pass through the suppression pool before reaching the outer containment region, sprays are not nearly as important. This is, in part, because the source term has already been reduced and, in part, because the decontamination factors for suppression pools and containment sprays are not multiplicative since they selectively remove similar-sized aerosols.

6.5 Offsite Consequence Results

Figure 6.8 displays the frequency distributions in the form of graphical plots of the complementary cumulative distribution functions (CCDFs) of four offsite consequence measures—early fatalities, latent cancer fatalities, and the 50-mile and the entire site region population exposures (in person-rems). These CCDFs include contributions from all source terms associated with reactor accidents caused by internal initiating events. Four CCDFs,

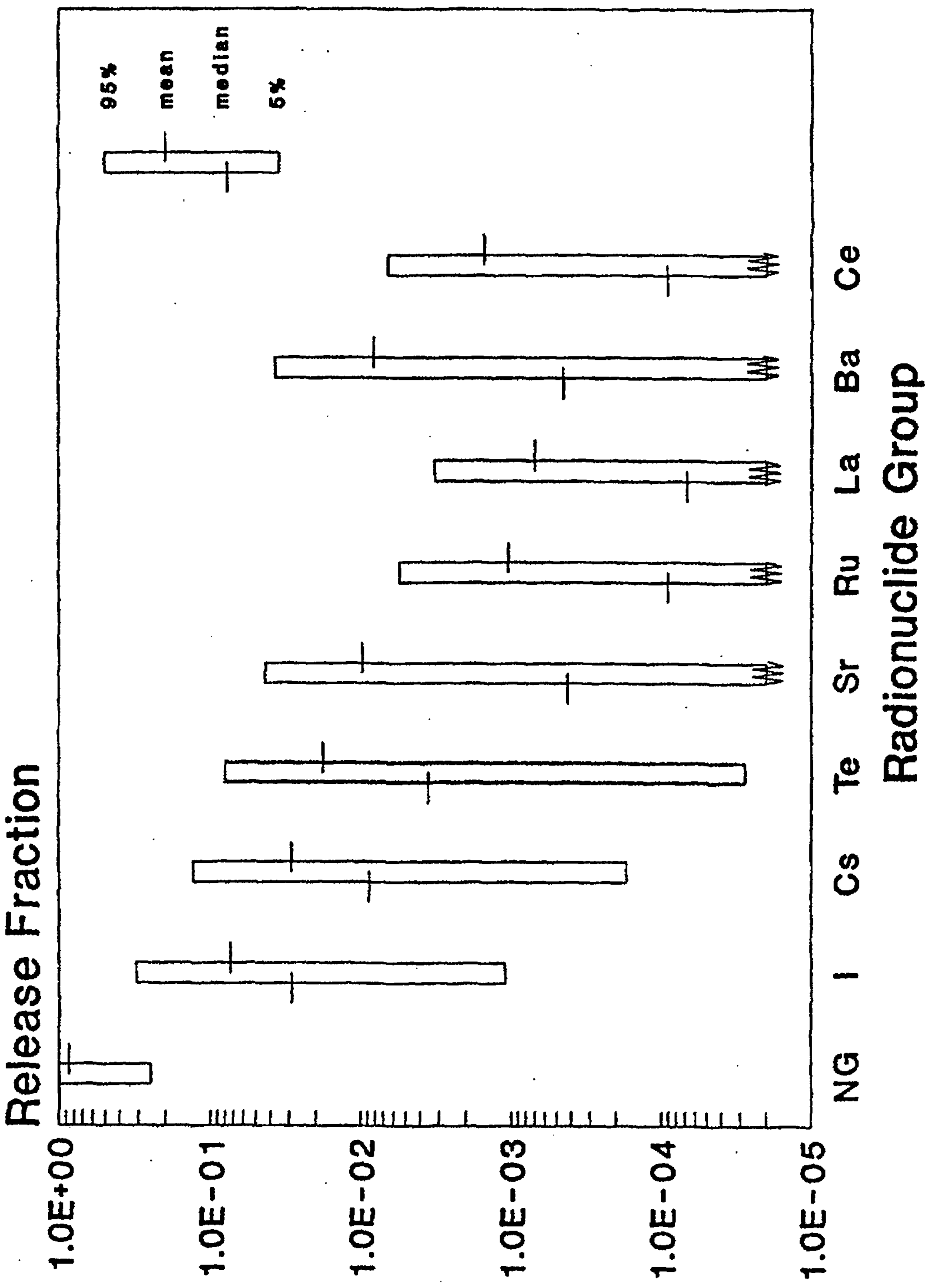


Figure 6.6 Source term distributions for early containment failure with drywell failed and sprays unavailable at Grand Gulf.

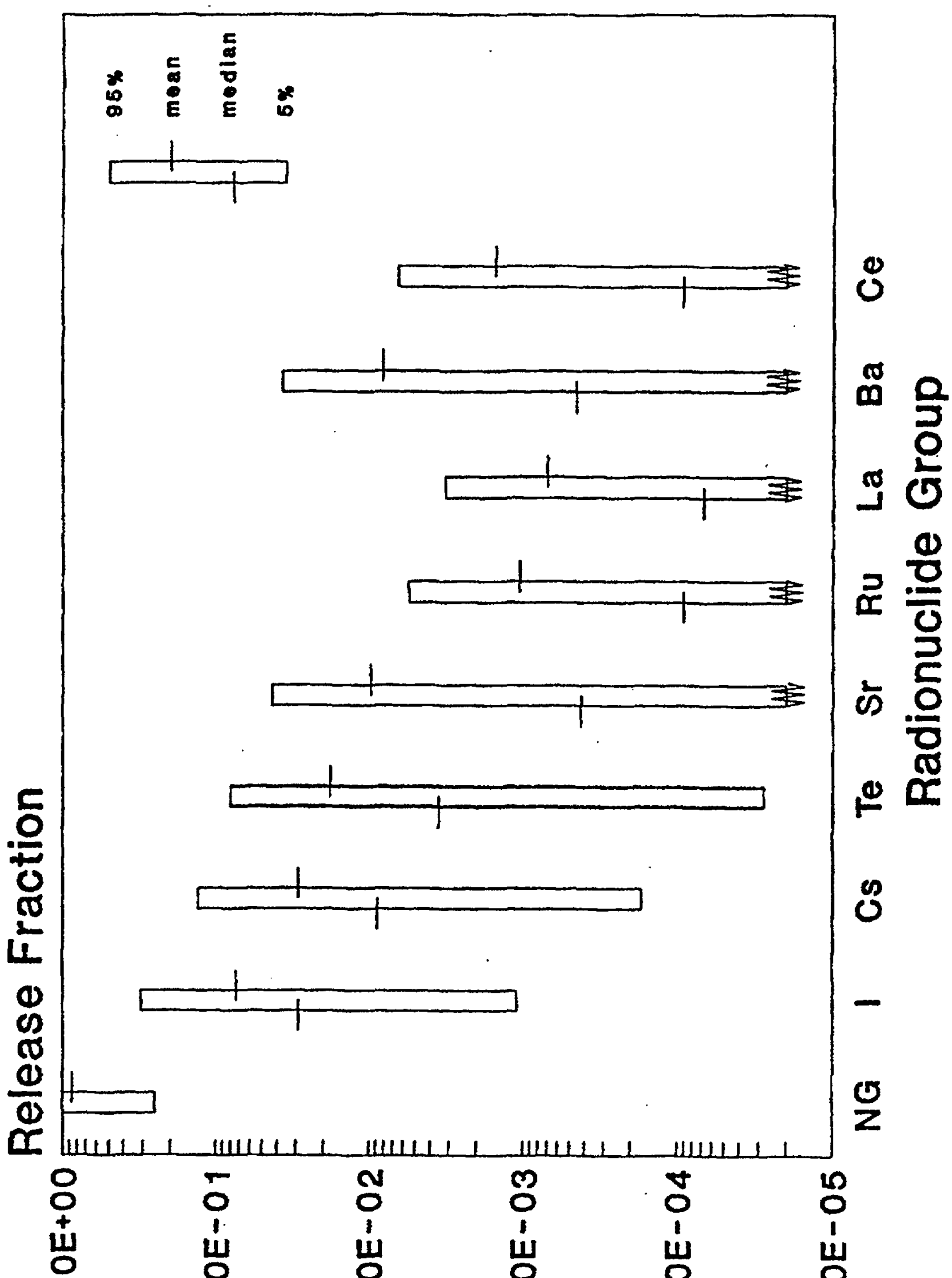
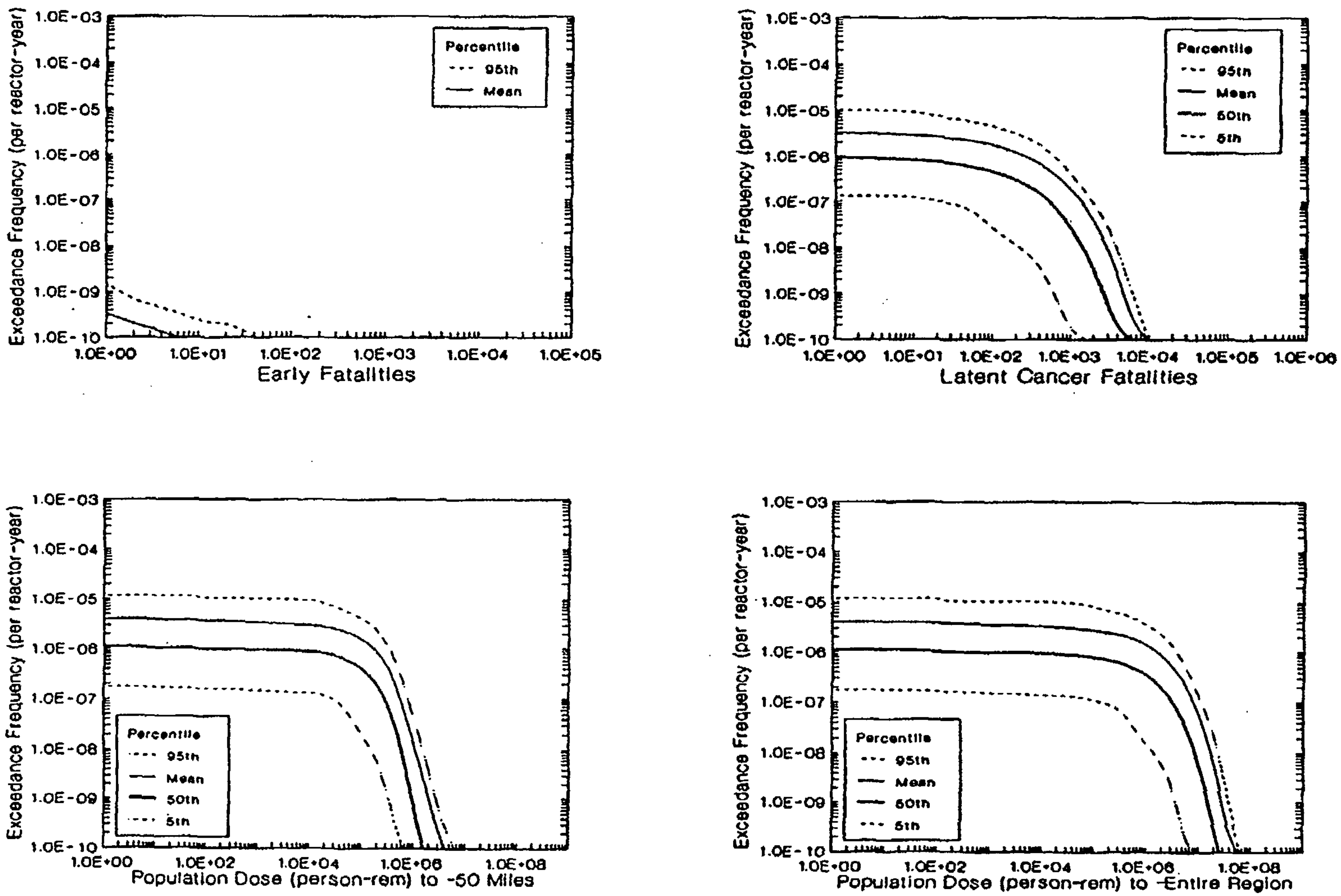


Figure 6.7 Source term distributions for early containment failure with drywell intact at Grand Gulf.



Note: As discussed in Reference 6.4, estimated consequences at frequencies 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 6.8 Frequency distributions of offsite consequence measures at Grand Gulf (internal initiators).

namely, the 5th percentile, 50th percentile (median), 95th percentile, and the mean CCDFs, are shown for each consequence measure.

Grand Gulf plant-specific and site-specific parameters were used in the consequence analyis for these CCDFs. The plant-specific parameters included source terms and their frequencies, the licensed thermal power (3833 MWt) of the reactor, and the approximate physical dimensions of the power plant building complex. The site-specific parameters included exclusion area radius (696 meters), meteorological data for 1 full year collected at the meteorological tower, the site region population distribution based on the 1980 census data, topography (fraction of the area that is land—the remaining fraction is assumed to be water), land use, agricultural practice and productivity, and other economic data for up to 1,000 miles from the Grand Gulf plant.

The consequence estimates displayed in these figures have incorporated the benefits of the following protective measures: (1) evacuation of 99.5 percent of the population within the 10-mile plume exposure pathway emergency planning zone (EPZ), (2) early relocation of the remaining population only from the heavily contaminated areas both within and outside the 10-mile EPZ, and (3) decontamination, temporary interdiction, or condemnation of land, property, and foods contaminated above acceptable levels.

The population density within the Grand Gulf 10-mile EPZ is about 30 persons per square mile. The average delay time before evacuation (after a warning prior to radionuclide release) from the 10-mile EPZ and average effective evacuation speed used in the analyses were derived from information contained in a utility-sponsored Grand Gulf evacuation time estimate study (Ref. 6.6) and the NRC requirements for emergency planning.

The results displayed in Figure 6.8 are discussed in Chapter 11.

6.6 Public Risk Estimates

6.6.1 Results of Public Risk Estimates

A detailed description of the results of the Grand Gulf risk analysis is provided in Reference 6.2. For this summary report, results are provided for the following measures of public risk:

- Early fatality risk,
- Latent cancer fatality risk,
- Population dose within 50 miles of the site,
- Population dose within the entire site region,
- Individual early fatality risk in the population within 1 mile of the Grand Gulf exclusion area boundary, and
- Individual latent cancer fatality risk in the population within 10 miles of the Grand Gulf site.

The first four of the above measures are commonly used measures in nuclear power plant risk studies. The last two are those used to compare with the NRC safety goals (Ref. 6.7).

The results of the Grand Gulf risk studies using the above measures are shown in Figures 6.9 through 6.11. The figures display the variabilities in mean risks estimated from meteorology-averaged conditional mean values of the consequence measures. In comparison to the risks from the other plants in this study, Grand Gulf has the lowest risk estimates. The results are much below those of the Reactor Safety Study (Ref. 6.8). The individual early and latent cancer fatality risks are far below the NRC safety goals. Details of the comparison of results are provided in Chapter 12.

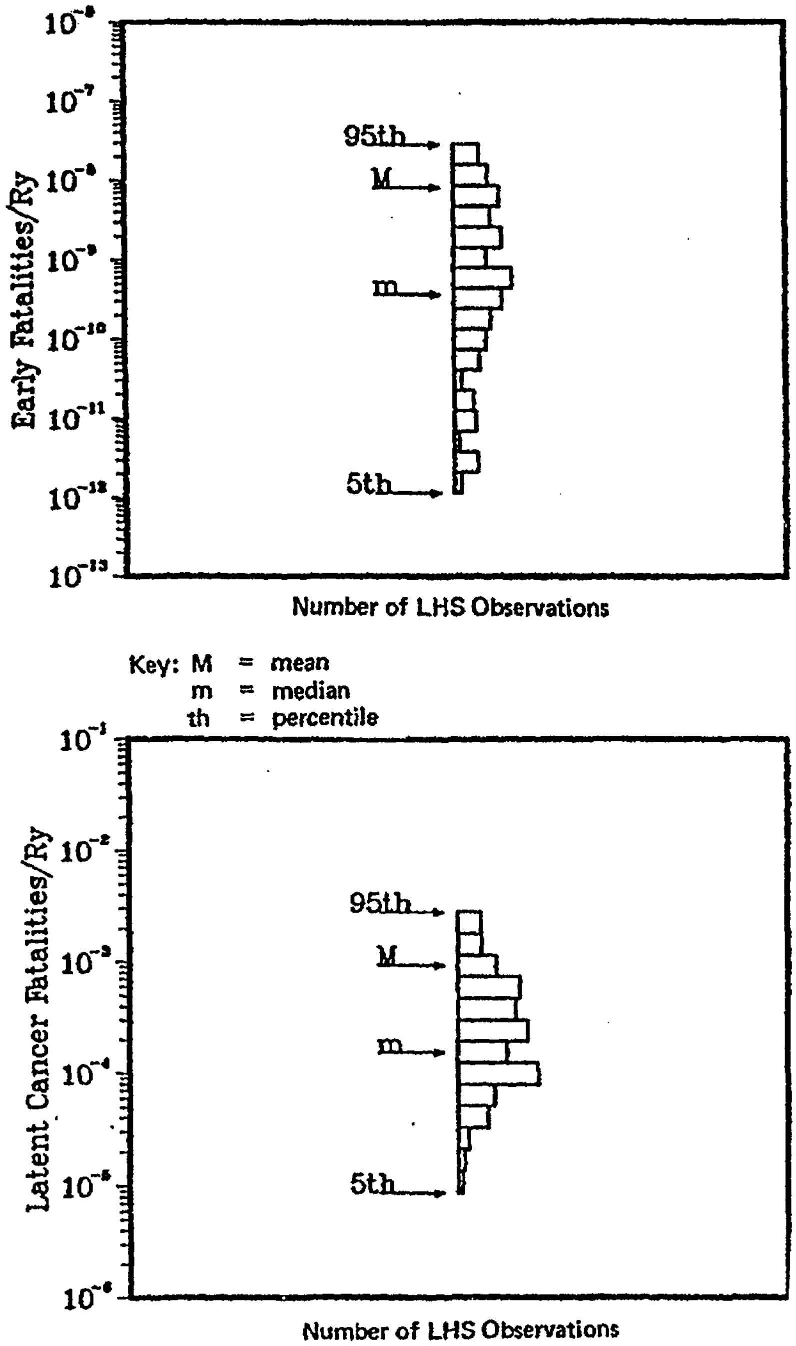
The results in Figure 6.9 have been analyzed to identify the relative contributions of accident sequences and containment failure modes to mean risk. These results are presented in Figures 6.12 and 6.13. As may be seen, the mean early fatality risk at Grand Gulf is dominated by short-term station blackout sequences. The majority of early fatality risk is associated with the coincidence of early containment failure and early suppression pool bypass.

The mean latent cancer fatality risk is also dominated by the short-term station blackout group. The major contributors to risk are from (1) early containment and early suppression pool bypass, and (2) late containment failure.

6.6.2 Important Plant Characteristics (Risk)

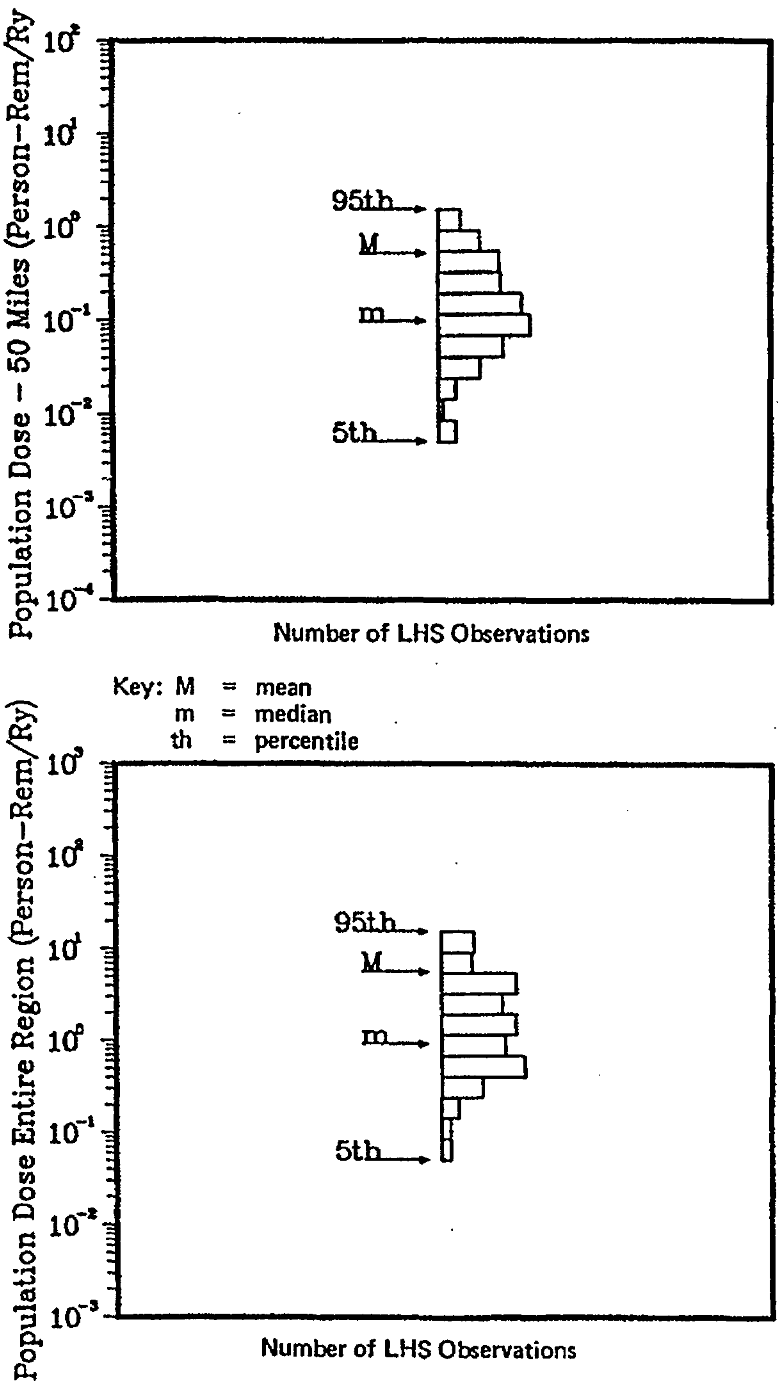
As mentioned before, risk to the public from the operation of the Grand Gulf plant is lower than the other four plants in this study. Some of the plant features that contribute to these low risk estimates are described below.

- The very low early fatality risk at Grand Gulf is due to a combination of low core damage



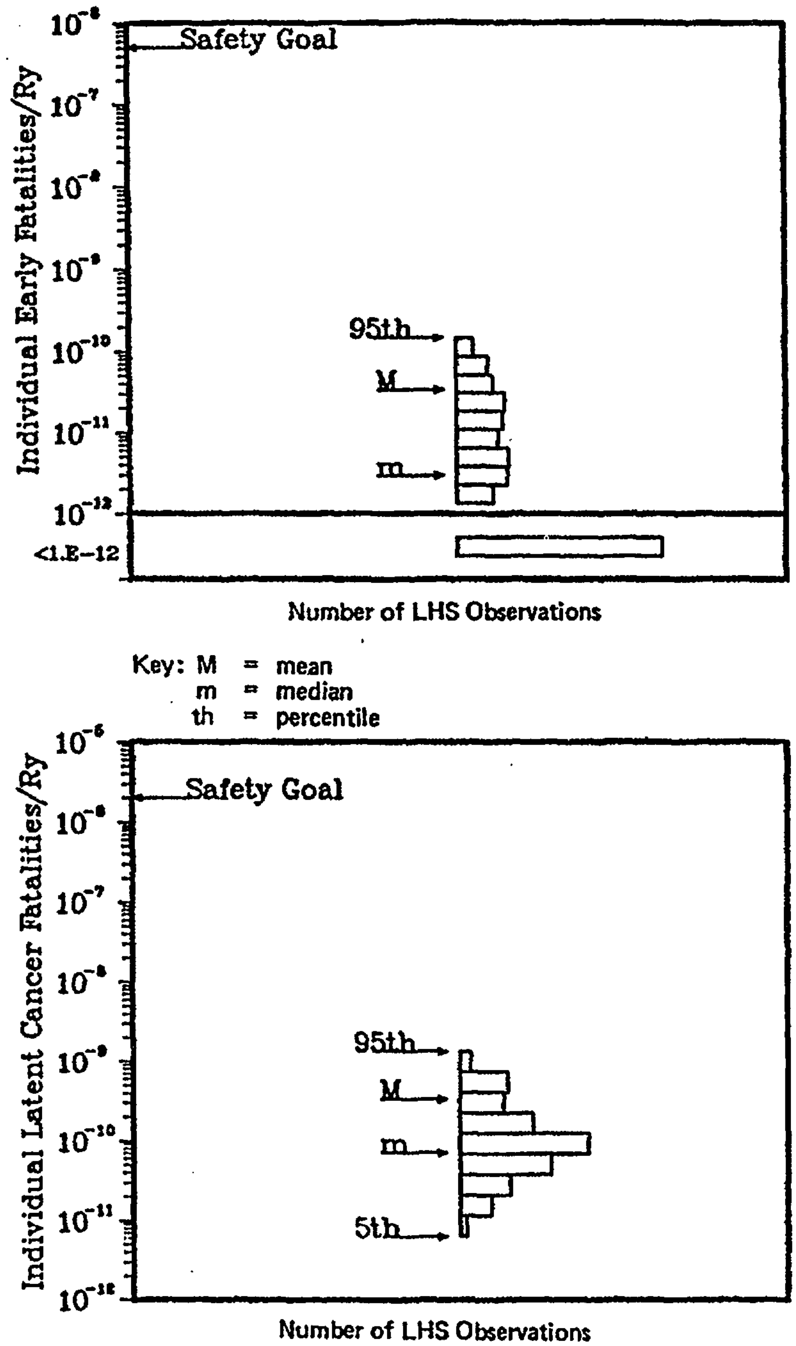
Note: As discussed in Reference 6.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 6.9 Early and latent cancer fatality risks at Grand Gulf (internal initiators).



Note: As discussed in Reference 6.4, 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 6.10 Population dose risks at Grand Gulf (internal initiators).



Note: As discussed in Reference 6.4, 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 6.11 Individual early and latent cancer fatality risks at Grand Gulf (internal initiators).

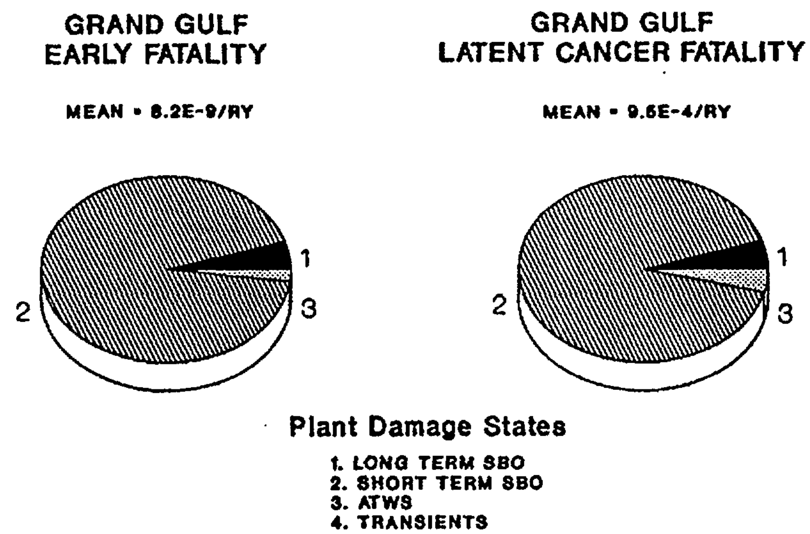


Figure 6.12 Major contributors (plant damage states) to mean early and latent cancer fatality risks at Grand Gulf (internal initiators).

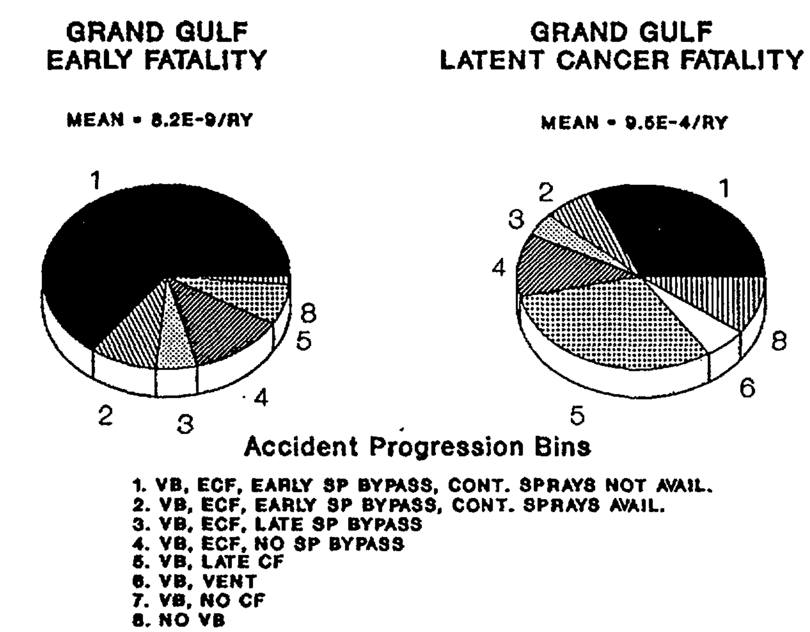


Figure 6.13 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Grand Gulf (internal initiators).

frequency, reduced source terms (as a result of suppression pool scrubbing), and low population density around the plant. The latter leads to short evacuation delays and fast evacuation speeds. Timing is not as important for latent cancer fatalities.

- Although the Grand Gulf plant has relatively high probability of early containment failure, caused mainly by hydrogen deflagration, the probability of early drywell failure, which may lead to a large source term, is about half of the probability of early containment failure. Furthermore, in most cases, in-vessel releases pass through the suppression pool.

- There is a high probability of having water in the reactor cavity following vessel breach. Thus, there is a high probability that core debris would be coolable. Even when any core-concrete interaction may occur, it is generally under water, and, therefore, the resulting releases are scrubbed by overlaying water (if not by the suppression pool).

REFERENCES FOR CHAPTER 6

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7. ZION PLANT RESULTS

7.1 Summary Design Information

The Zion Nuclear Plant is a two-unit site. Each unit is a four-loop Westinghouse nuclear steam supply system rated at 1100 MWe and is housed in a large, prestressed concrete, steel-lined dry containment. The balance of plant systems were engineered by Sargent & Lundy. Located on the shore of Lake Michigan, about 40 miles north of Chicago, Illinois, Zion 1 started commercial operation in December 1973. Some important design features of the Zion plant are described in Table 7.1. A general plant schematic is provided in Figure 7.1.

This chapter provides a summary of the results provided in the risk analyses underlying this report (Refs. 7.1 and 7.2). A discussion of perspectives with respect to these results is provided in Chapters 8 through 12.

7.2 Core Damage Frequency Estimates

7.2.1 Summary of Core Damage Frequency Estimates*

The core damage frequency and risk analyses performed for this study considered accidents initiated only by internal events (Ref. 7.1); no external-event analyses were performed. The core damage frequency results obtained are provided in tabular form in Table 7.2. This study calculated a total median core damage frequency from internal events of 2.4E–4 per year.

7.2.1.1 Zion Analysis Approach

The Zion plant was previously analyzed in the Zion Probabilistic Safety Study (ZPSS), performed by the Commonwealth Edison Company, and in the review and evaluation of the ZPSS (Ref. 7.3), commonly called the Zion Review prepared by Sandia National Laboratories.

Since previous analyses of Zion already existed, it was decided to perform an update of the previous analyses rather than perform a complete reanalysis. Therefore, this analysis of Zion represents a limited rebaseline and extension of the dominant accident sequences from the ZPSS in light of the Zion Review comments, although incorporating some methods and issues (such as common-cause failure treatment, electric power recovery, and reactor coolant pump seal LOCA modeling) used in the other four plant studies.

*In general, the results and perspectives provided here do not reflect recent modifications to the Zion plant. The benefit of the changes is noted, however, in specific places in the text (and discussed in more detail in Section 15 of Appendix C).

The objective of this study was to perform an analysis that updated the previous Zion analyses and cast the model in a manner more consistent with the other accident frequency analyses. The models were not completely reconstructed in the small-event-tree, large-fault-tree modeling method used in the study of the other NUREG–1150 plants. Instead, the small-fault-tree, large-event-tree models from the original ZPSS were used as the basis for the update. These models were then revised according to the comments from Reference 7.3 and were enhanced to address risk issues using methods employed by the other plant studies.

This study incorporated specific issues into the systems and accident sequence models of the ZPSS. These issues reflect both changes in the Zion plant and general PRA assumptions that have arisen since the ZPSS was performed. New dominant accident sequences were determined by modifying and requantifying the event tree models developed for ZPSS. The major changes reflect the need for component cooling water and service water for emergency core cooling equipment and reactor coolant pump seal integrity. The original set of plant-specific data used in the ZPSS and Zion Review was verified as still valid and was used for this study. Additional discussion of the Zion methods is provided in Appendix A.

7.2.1.2 Internally Initiated Accident Sequences

A detailed description of accident sequences important at the Zion plant is provided in Reference 7.1. For this summary report, the accident sequences described in that reference have been grouped into six summary plant damage states. These are:

- Station blackout,
- Loss-of-coolant accident (LOCA),
- Component cooling water and service water induced reactor coolant pump seal LOCAs,
- Anticipated transients without scram (ATWS),

Table 7.1 **Summary of design features: Zion Unit 1.**

1. High-Pressure Injection	a. Two centrifugal charging pumps. b. Two 1500-psig safety injection pumps. c. Charging pumps inject through boron injection tank. d. Provides seal injection flow. e. Requires component cooling water.
2. Low-Pressure Injection	a. Two RHR pumps deliver flow when RCS is below about 170 psig. b. Heat exchangers downstream of pumps provide recirculation heat removal. c. Recirculation mode takes suction on containment sump and discharges to the RCS, HPI suction, and/or containment spray pump suction. d. Pumps and heat exchangers require component cooling water.
3. Auxiliary Feedwater	a. Two 50 percent motor-driven pumps and one 100 percent turbine-driven pump. b. Pumps take suction from own unit condensate storage tank (CST) but can be manually crosstied to the other unit's CST.
4. Emergency Power System	a. Each unit consists of three 4160 VAC class 1E buses, each feeding one 480 VAC class 1E bus and motor control center. b. For the two units there are 5 diesel generators, with one being a swing diesel generator shared by both units. c. Three trains of dc power are supplied from the inverters and 3 unit batteries.
5. Component Cooling Water	a. Shared system between both units. b. Consists of 5 pumps, 3 heat exchangers, and 2 surge tanks. c. Cools RHR heat exchangers, RCP motors and thermal barriers, RHR pumps, SI pumps, and charging pumps. d. One of 5 pumps can provide sufficient flow.
6. Service Water	a. Shared system between both units. b. Consists of 6 pumps and 2 supply headers. c. Cools component cooling heat exchangers, containment fan coolers, diesel generator coolers, auxiliary feedwater pumps. d. Two of 6 pumps can supply sufficient flow.
7. Containment Structure	a Large, dry, prestressed concrete. b. 2.6 million cubic foot volume. c. 49 psig design pressure.
8. Containment Spray	a. Two motor-driven pumps and 1 independent diesel-driven pump. b. No train crossties. c. Water supplied by refueling water storage tank.
9. Containment Fan Coolers	a. Five fan cooler units, a minimum of 3 needed for post-accident heat removal. b. Fan units shift to low speed on SI signal. c. Coolers require service water.

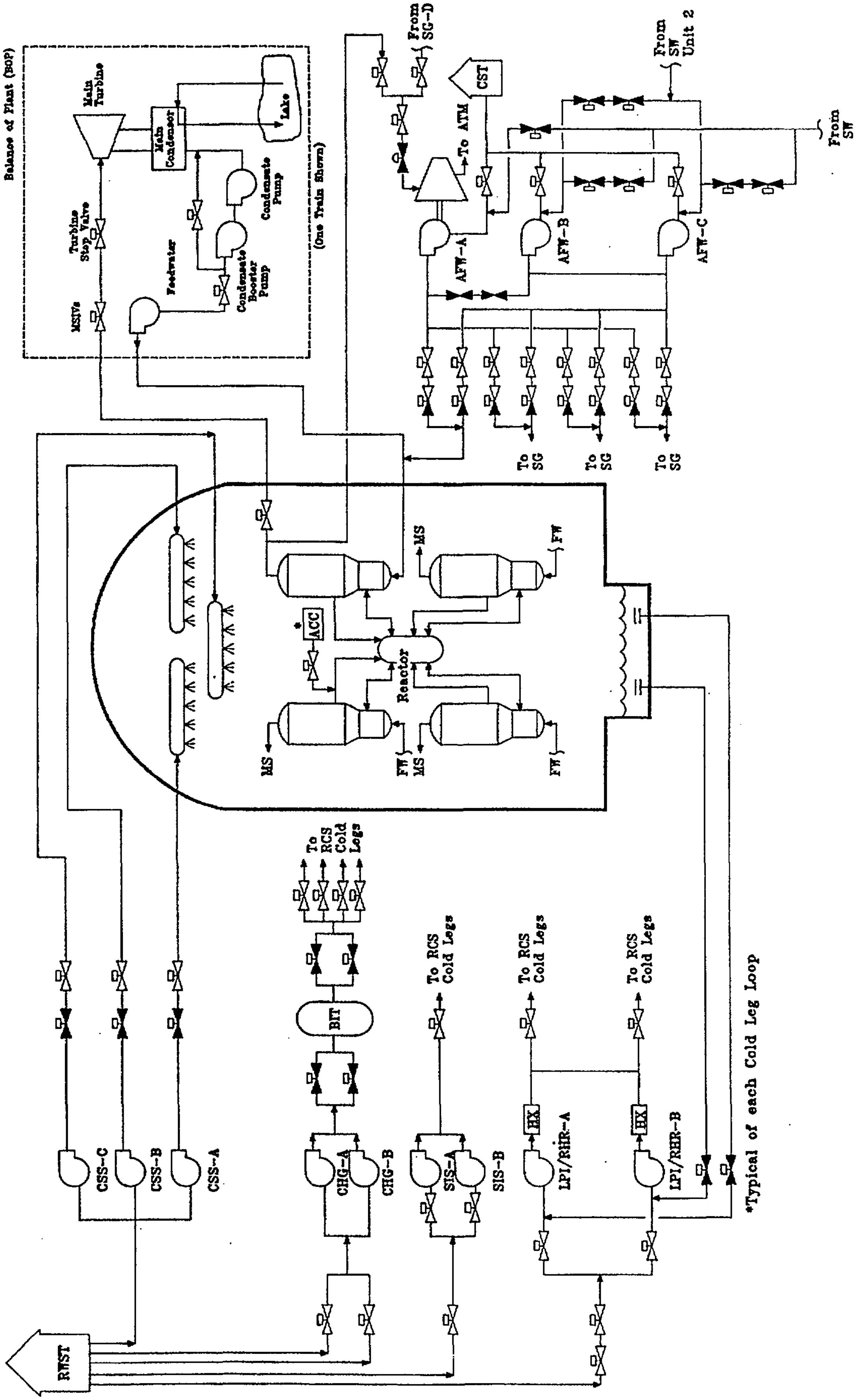


Figure 7.1 Zion plant schematic.

Table 7.2 Summary of core damage frequency results: Zion.

	5%	Median	Mean	95%
Internal Events	1.1E–4	2.4E–4	3.4E–4*	8.4E–4

*See text (Section 7.2.1) for benefit of recent modifications.

- Interfacing-system LOCA and steam generator tube rupture (SGTR), and
- Transients other than station blackout and ATWS.

The relative contribution of the accident types to mean core damage frequency at Zion is shown in Figure 7.2. It is seen that the dominating contributors to the core damage frequency are the loss of component cooling water and loss of service water. The more probable combinations of failures are:

- Reactor coolant pump seals fail because of the loss of cooling and injection. Core damage occurs because of failure to recover the service water/component cooling water systems in time to reestablish reactor coolant system inventory control. In cases with failure of the service water system, containment fan coolers are also failed.
- Reactor coolant pump seals fail because of the loss of cooling and injection. The cooling system is recovered in time to provide injection from the refueling water storage tank (RWST). Recirculation cooling fails to continue to provide long-term inventory control.

To address the issue of the importance of component cooling water system failures, Commonwealth Edison (the Zion licensee) committed in 1989 to perform the following actions (Ref. 7.4):

- Provide an auxiliary water supply to each charging pump's oil cooler via either the service water system or fire protection system. Hoses, fittings, and tools will be maintained locally at each unit's charging pump area allowing for immediate hookup to existing taps on the oil coolers, if required. As an interim measure, a standing order in the control room will instruct operators as to how and when to hook up auxiliary water to the oil coolers.
- Formal procedures, including a 10 CFR 50.59 review addressing the loss of component cooling water system scenario, will be fully implemented within 60 days (of the date of Ref. 7.4) to supersede the standing order.
- When new heat-resistant reactor coolant pump seal o-rings are made available by Westinghouse, the existing o-rings will be changed when each pump is disassembled for routine scheduled seal maintenance.

These actions provide a backup water source to the Zion station charging pump oil coolers.

As of October 1990, Commonwealth Edison had performed some of the noted actions (Ref. 7.5). Sensitivity studies have been performed to assess the benefit of the modifications made to date. These studies, discussed in more detail in Section C.15 of Appendix C, indicate that the Zion estimated mean core damage frequency has been reduced from 3.4E–4 per year to approximately 6E–5 per year.

7.2.2 Important Plant Characteristics (Core Damage Frequency)

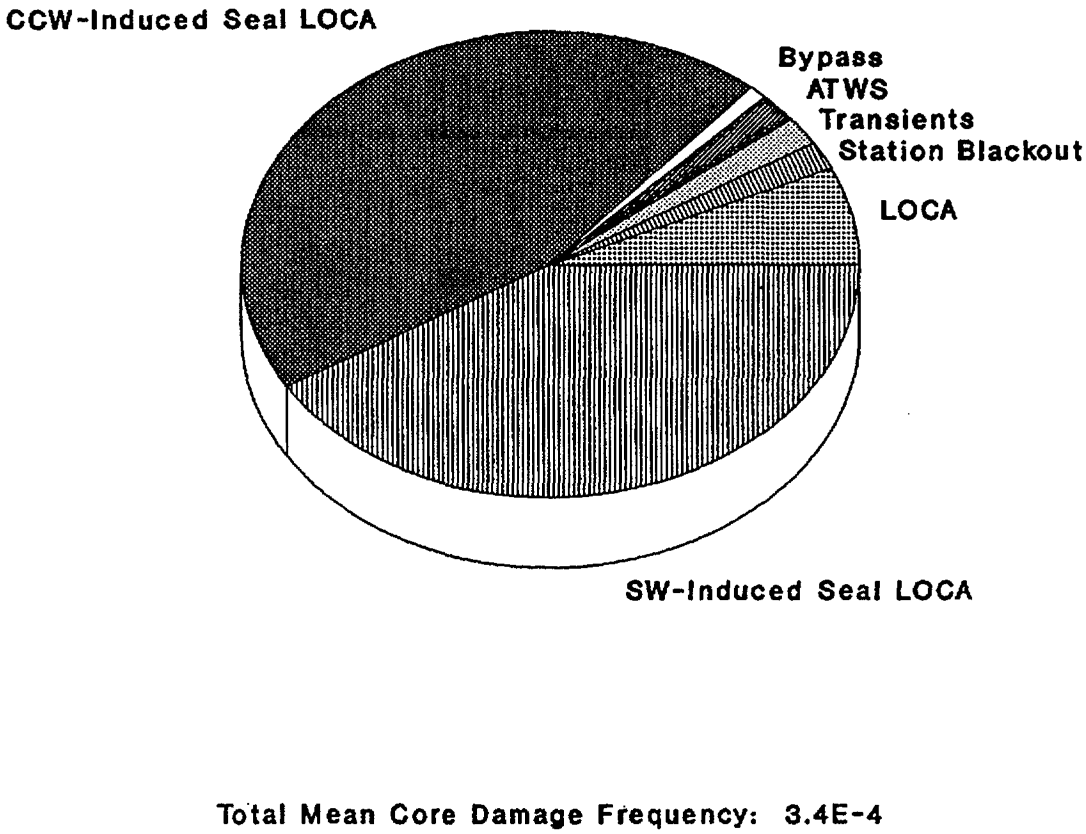
Characteristics of the Zion plant design and operation that have been found to be important in the analysis of the core damage frequency include:

1. **Shared Systems Between Units**

   The Zion nuclear station shares the service water and component cooling water (CCW) systems between the two units. Power is supplied to these systems from all five onsite diesel generators.

2. **Crossties Between Units**

   Crossties between units exist for the condensate storage tanks to provide water supply for the auxiliary feedwater system. Crossties also exist between Unit 1 and Unit 2 ac power systems, as well as between Unit 1 and Unit 2 dc power systems.



Note: See text (Section 7.2.1) for benefit of recent modifications.

Figure 7.2 Contributors to mean core damage frequency from internal events at Zion.

3. **Diesel Generators**

Zion is a two-unit site with five emergency diesel generators. One diesel generator is a swing diesel that can be lined up to supply either unit. This differs from a number of other two-unit sites that have only four diesel generators on site. The Zion diesel generators are dependent on a common service water system for sustained operation.

4. **Support System Dependencies**

The component cooling water system supplies cooling water for the reactor coolant pump thermal barriers and for the charging pumps that supply seal injection. Failure of the component cooling water system results in a major challenge to reactor coolant pump seal integrity. In addition, failure of the component cooling water support systems (service water and ac power) also leads to loss of reactor coolant pump seal integrity. In contrast, some other PWRs do not have a common dependency for both seal cooling and seal injection; therefore, at other PWRs, seal LOCAs are only important in station blackout cases. As indicated above, the licensee has committed to and implemented plant changes to reduce this dependency.

5. **Battery Depletion Time**

The battery depletion time following a complete loss of all ac power was estimated at 6 hours, somewhat longer than that found at some other plants. The additional time tends to reduce the significance of the station blackout sequences as contributors to the core damage frequency.

6. **Reactor Coolant Pump Seal Performance**

   The inability of the reactor coolant pump seals to survive loss of cooling and injection without developing significant leakage dominates the core damage frequency. As noted above, the licensee has committed to replacing present seals with a new model.

7.2.3 Important Operator Actions

Several operator actions and recovery actions are important to the analysis of the core damage frequency. While the analysis included a wide range of operator actions from test and maintenance errors before an initiating event to recovery actions well into an accident sequence, the following actions surface as the most important:

- Successful switchover to recirculation

  The operator must recognize that switchover should be initiated, take action to open the proper set of motor-operated valves depending on reactor coolant system conditions, and verify that recirculation flow is proper.

- Successful execution of feed and bleed cooling

  The operator must recognize that secondary cooling is lost, establish sufficient injection flow, open both power-operated relief valves (and their block valves, if necessary), and verify that adequate heat removal is taking place.

- Recovery of the component cooling water and service water systems

  The operator must recognize that the failure of equipment or rising equipment operating temperatures are due to failure of the service water or component cooling water systems, determine the cause of system failure, and take appropriate action to isolate ruptures, restart pumps, and provide alternative cooling paths as required by the situation.

- Actions to refill the RWST in the event of recirculation failure

  This action requires that the operator recognize the failure of recirculation cooling in sufficient time that refill can begin before core damage occurs. The operator must then carry out the procedure for emergency refill of the RWST. This action is not adequate for inventory control in the case of larger LOCAs because of the limitations of the refilling equipment.

Switchover to recirculation cooling and initiation of feed and bleed cooling were included in the original Zion Probabilistic Safety Study and have been given close scrutiny by the licensee. Each one of these actions is present in the emergency procedures. Appropriate consideration of the procedures, scenarios, timing, and training went into the determination of the human error probabilities associated with these actions. Because of the importance and uncertainty associated with several of these actions, they were addressed in the sensitivity analyses. However, the refilling of the RWST in the event of recirculation failure and recovery of CCW and service water were not included in the original Zion Probabilistic Safety Study. Appropriate consideration of the procedures, scenarios, timing, and training went into the determination of the human error probabilities associated with these actions. Because of the importance and uncertainty associated with several of these actions, they were addressed in the sensitivity analyses.

7.3 Containment Performance Analysis

7.3.1 Results of Containment Performance Analysis

The Zion containment consists of a large, dry containment building that houses the reactor pressure vessel, reactor coolant system piping, and the secondary system's steam generators. The containment building is a prestressed concrete structure with a steel liner. This building has a volume of 2.6 million cubic feet with a design pressure of 49 psig and an estimated mean failure pressure of 150 psia. The principal threats to containment integrity from potential severe accident sequences are steam explosions, overpressurization from direct containment heating effects, bypass events, and isolation failures. As previously discussed in Chapter 2, the methods used to estimate loads and containment structural response for Zion made extensive use of expert judgment to interpret and supplement the limited data (Ref. 7.2).

The results of the Zion containment analysis are summarized in Figures 7.3 and 7.4. Figure 7.3 displays information in which the conditional probabilities of four accident progression bins, e.g., early containment failure, are presented for each of five plant damage states, e.g., LOCA. This information indicates that, on a plant damage

ACCIDENT PROGRESSION BIN	PLANT DAMAGE STATE (Mean Core Damage Frequency)				
	SBO (9.34E-6)	LOCAs (3.14E-4)	Transients (1.36E-5)	V & SGTR (2.59E-7)	All (3.38E-4)
Early CF	0.025	0.014	0.012		0.014
Late CF	0.320	0.250	0.190		0.240
Bypass	0.001		0.004	1.000	0.007
No CF	0.660	0.740	0.790		0.730

Key: CF = Containment Failure

Figure 7.3 Conditional probability of accident progression bins at Zion.

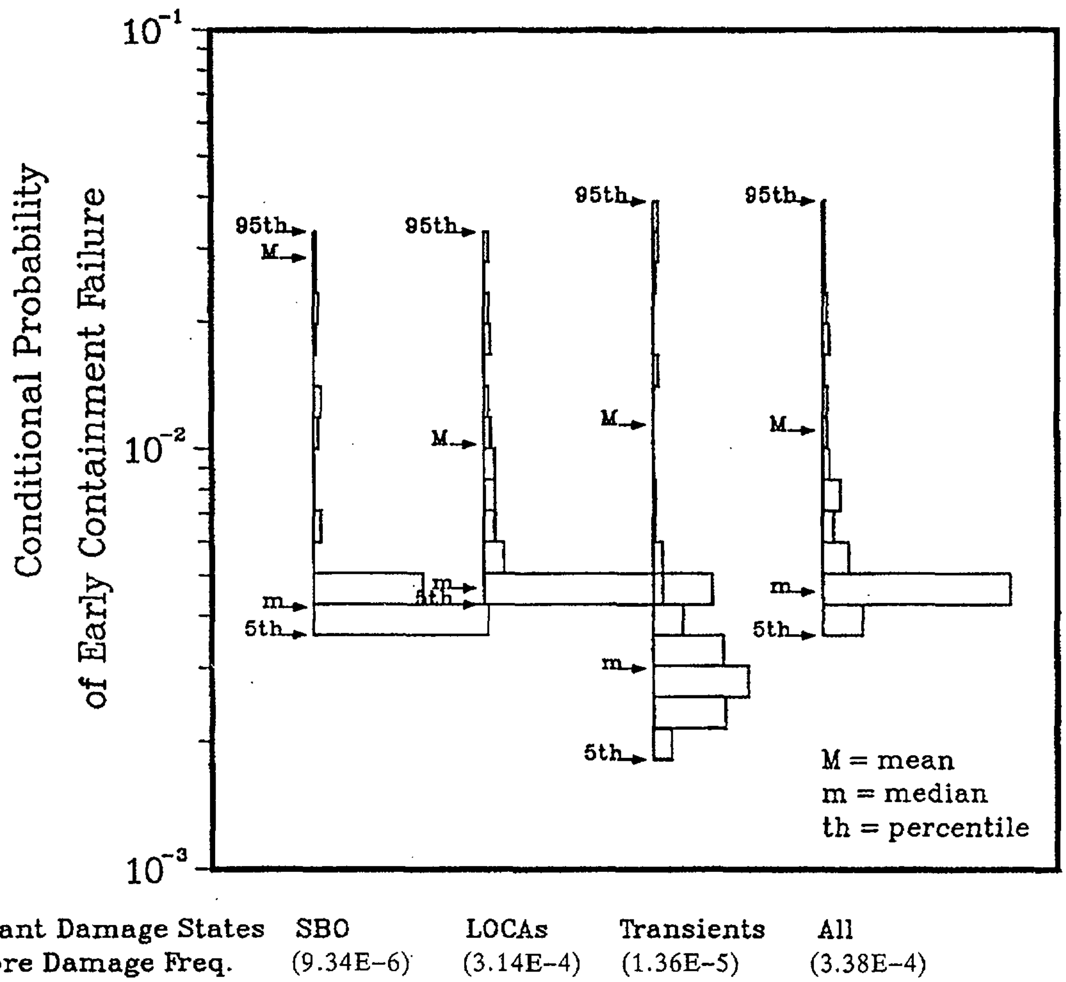


Figure 7.4 Conditional probability distributions for early containment failure at Zion.

state frequency-weighted average,* the mean conditional probabilities from internal events of (1) early containment failure from a combination of in-vessel steam explosions, overpressurization, and containment isolation failures is 0.014, (2) late containment failure, mainly from basemat meltthrough is 0.24, (3) containment bypass from interfacing-system LOCA and induced steam generator tube rupture (SGTR) is 0.006, and (4) probability of no containment failure is 0.73. Figure 7.4 further displays the conditional probability distributions of early containment failure for the plant damage states, thereby providing the estimated range of uncertainties in these containment failure predictions. The principal conclusion to be drawn from the information in Figures 7.3 and 7.4 is that the probability of early containment failure for Zion is low, i.e., 1 to 2 percent.

Additional discussion on containment performance is provided in Chapter 9.

7.3.2 Important Plant Characteristics (Containment Performance)

Characteristics of the Zion design and operation that are important to containment performance include:

1. **Containment Volume and Pressure Capability**

   The combined magnitude of Zion's containment volume and estimated failure pressure provide considerable capability to withstand severe accident threats.

2. **Reactor Cavity Geometry**

   The Zion containment design arrangement has a large cavity directly beneath the reactor pressure vessel that communicates to the lower containment by means of an instrument tunnel. Provided the contents of the refueling water storage tank have been injected prior to vessel breach, this arrangement should provide a mechanism for quenching the molten core for some severe accidents (although there remains some uncertainties with respect to the coolability of molten core debris in such circumstances).

*Each value in the column in Figure 7.3 labeled "All" is a frequency-weighted average obtained by calculating the products of individual accident progression bin conditional probabilities for each plant damage state and the ratio of the frequency of that plant damage state to the total core damage frequency.

7.4 Source Term Analysis

7.4.1 Results of Source Term Analysis

The containment performance results for the Zion (large, dry containment) plant and the Surry (subatmospheric containment) plant are quite similar. The source terms for analogous accident progression bins are also quite similar. Figure 7.5 illustrates the source term for early containment failure. As at Surry, the source terms for early failure are somewhat less than those for containment bypass. Within the range of the uncertainty band, however, the source terms from early containment failure are potentially large enough to result in some early fatalities.

The most likely outcome of a severe accident at the Zion plant is that the containment would not fail. Figure 7.6 illustrates the range of source terms for the no containment failure accident progression bin. Other than for the noble gas and iodine radionuclide groups, the entire range of source terms is below a release fraction of 10E-5.

Additional discussion on source term perspectives is provided in Chapter 10.

7.4.2 Important Plant Characteristics (Source Term)

1. **Containment Spray System**

   The containment spray system at the Zion plant is not required to operate to provide long-term cooling to the containment, in contrast to the Surry plant. Operation of the spray system is very effective, however, in reducing the airborne concentration of aerosols. Other than the release of noble gases and some iodine evolution, the release of radioactive material to the atmosphere resulting from late containment leakage or basemat meltthrough in which sprays have operated for an extended time would be very small. The source terms for the late containment failure accident progression bin are slightly higher than, but similar to, those of the no containment failure bin illustrated in Figure 7.6.

2. **Cavity Configuration**

   The Zion cavity is referred to as a wet cavity, in that the accumulation of a relatively small amount of water on the containment floor will lead to overflow into the cavity. As a result, there is a substantial likelihood of eliminating by forming a coolable debris bed or

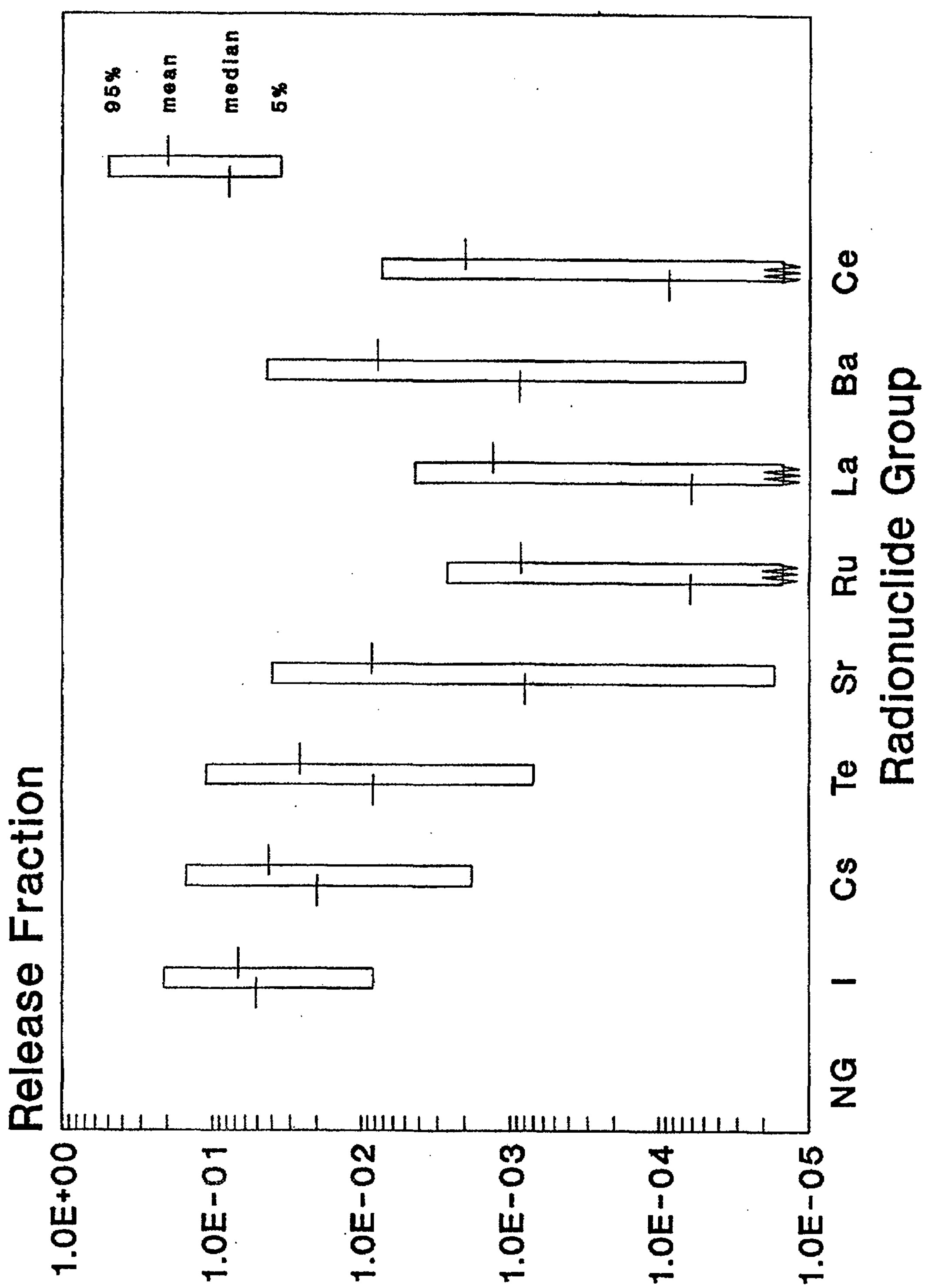


Figure 7.5 Source term distributions for early containment failure at Zion.

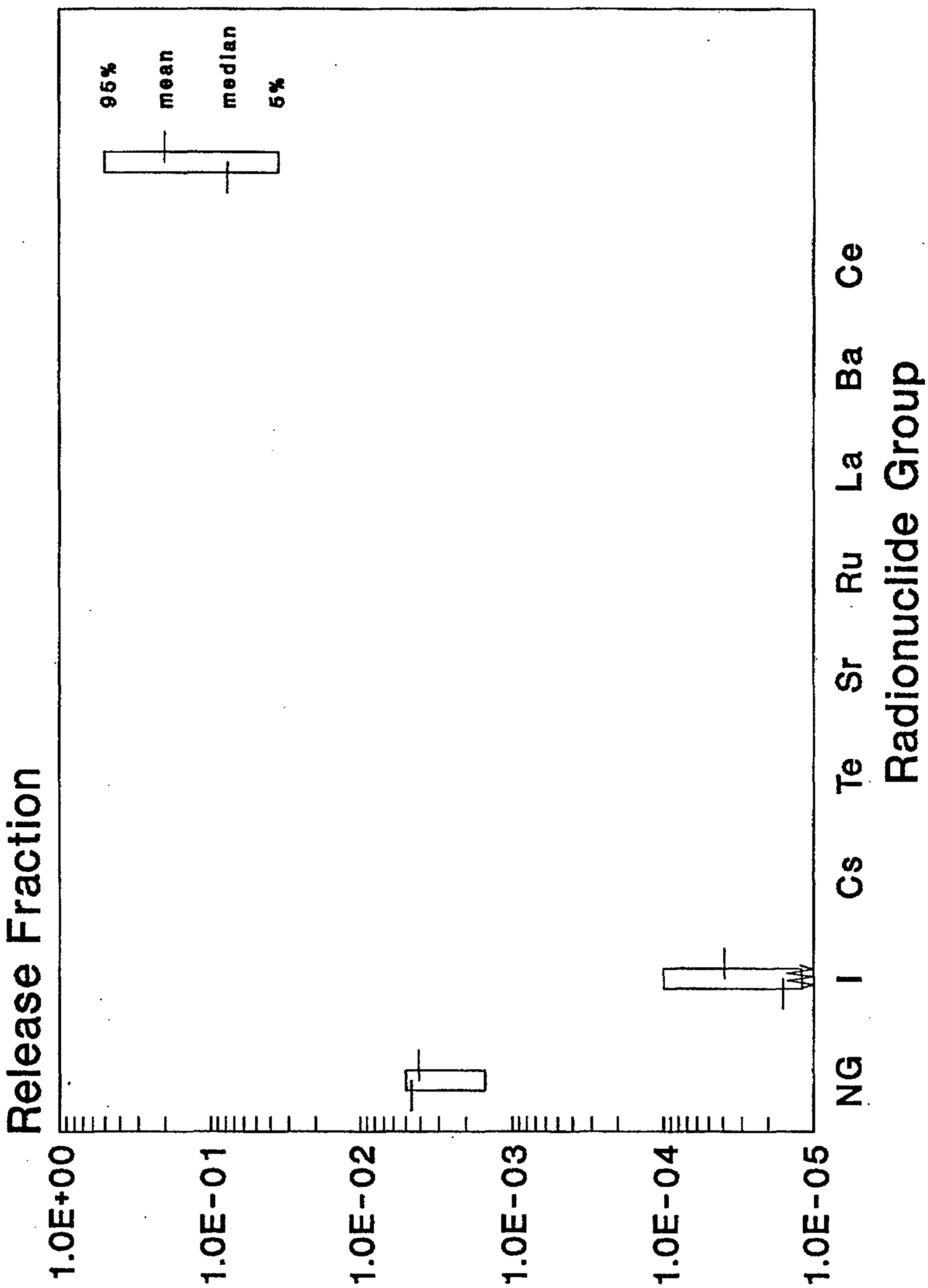


Figure 7.6 Source term distributions for no containment failure at Zion.

mitigating by the presence of an overlaying pool of water the release of radionuclides from core-concrete interactions.

7.5 Offsite Consequence Results

Figure 7.7 displays the frequency distributions in the form of graphical plots of the complementary cumulative distribution functions (CCDFs) of four offsite consequence measures—early fatalities, latent cancer fatalities, and the 50-mile region and entire site region population exposures (in person-rems). These CCDFs include contributions from all source terms associated with reactor accidents caused by internal initiating events. Four CCDFs, namely, the 5th percentile, 50th percentile (median), 95th percentile, and the mean CCDFs are shown for each consequence measure.

Zion plant-specific and site-specific parameters were used in the consequence analysis for these CCDFs. The plant-specific parameters included source terms and their frequencies, the licensed thermal power (3250 MWt) of the reactor, and the approximate physical dimensions of the power plant building complex. The site-specific parameters included exclusion area radius (400 meters), meteorological data for 1 full year collected at the site meteorological tower, the site region population distribution based on the 1980 census data, topography (fraction of the area which is land—the remaining fraction is assumed to be water), land use, agricultural practice and productivity, and other economic data for up to 1,000 miles from the Zion plant.

The consequence estimates displayed in these figures have incorporated the benefits of the following protective measures: (1) evacuation of 99.5 percent of the population within the 10-mile plume exposure pathway emergency planning zone (EPZ), (2) early relocation of the remaining population only from the heavily contaminated areas both within and outside the 10-mile EPZ, and (3) decontamination, temporary interdiction, or condemnation of land, property, and foods contaminated above acceptable levels.

The population density within the Zion 10-mile EPZ is about 1360 persons per square mile. About 45 percent of the 10-mile EPZ is water. The average delay time before evacuation (after a warning prior to radionuclide release) from the 10-mile EPZ and average effective evacuation speed used in the analyses were derived from information contained in a utility-sponsored Zion evacuation time estimate study (Ref. 7.7) and in an independent analysis by the Federal Emergency Management Agency (Ref. 7.8) and the NRC requirements for emergency planning.

The results displayed in Figure 7.7 are discussed in Chapter 11.

7.6 Public Risk Estimates

7.6.1 Results of Public Risk Estimates*

A detailed description of the results of the Zion risk analysis is provided in Reference 7.2. For this summary report, results are provided for the following measures of public risk:

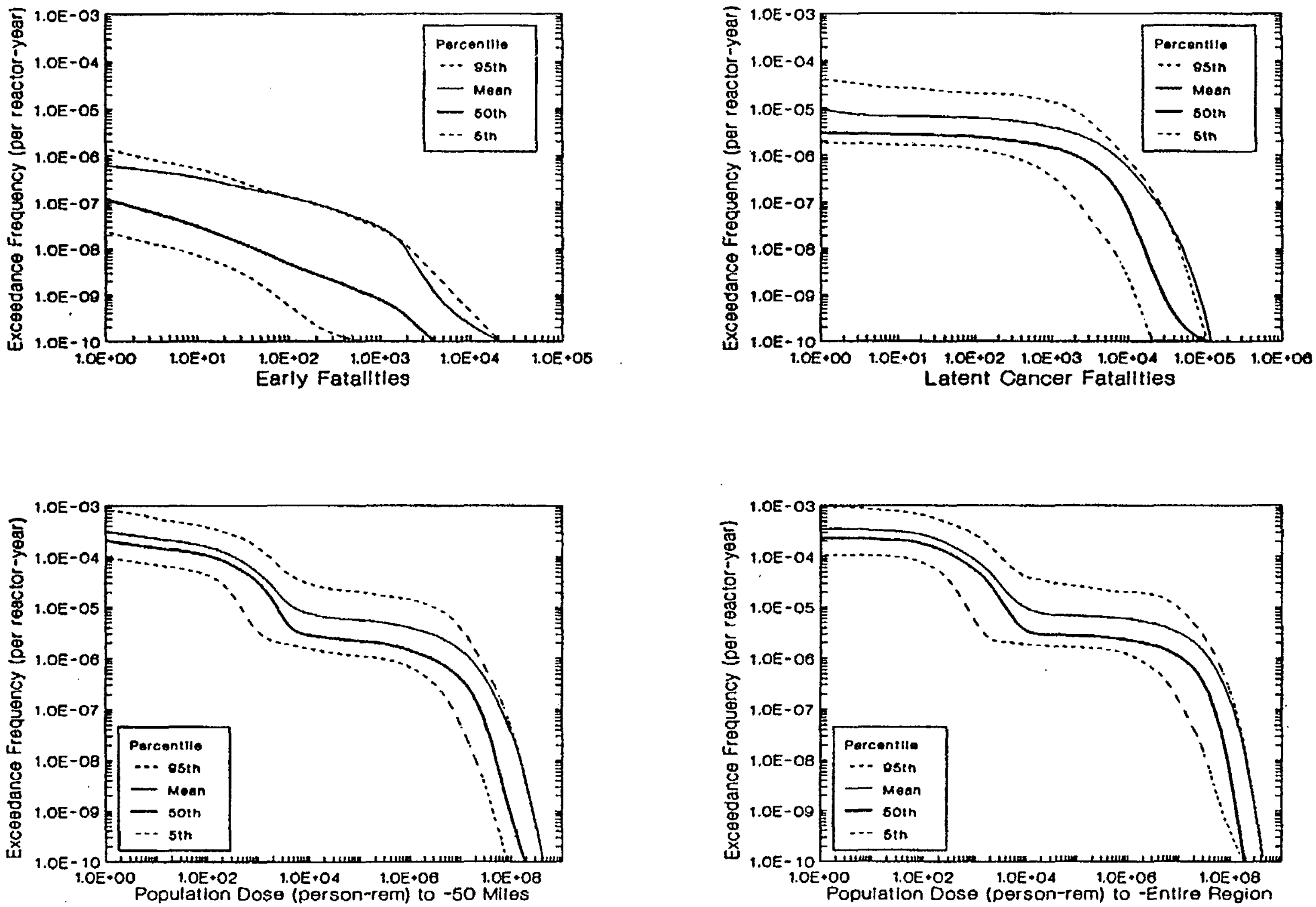
- Early fatality risk,
- Latent cancer fatality risk,
- Population dose within 50 miles of the site,
- Population dose within the entire site region,
- Individual early fatality risk in the population within 1 mile of the Zion exclusion area boundary, and
- Individual latent cancer fatality risk in the population within 10 miles of the Zion site.

The first four of the above measures are commonly used measures in nuclear plant risk studies. The last two are those used to compare with the NRC safety goals (Ref. 7.9).

The results of the Zion risk analyses are shown in Figures 7.8 through 7.10. The figures display variabilities in mean risks estimated from the meteorology-based conditional mean values of the consequence measures. The risk estimates are slightly higher than those of the other two PWR plants (Surry and Sequoyah) in this study. Individual early and latent cancer fatality risks are well below the NRC safety goals. Detailed comparisons of results are given in Chapter 12.

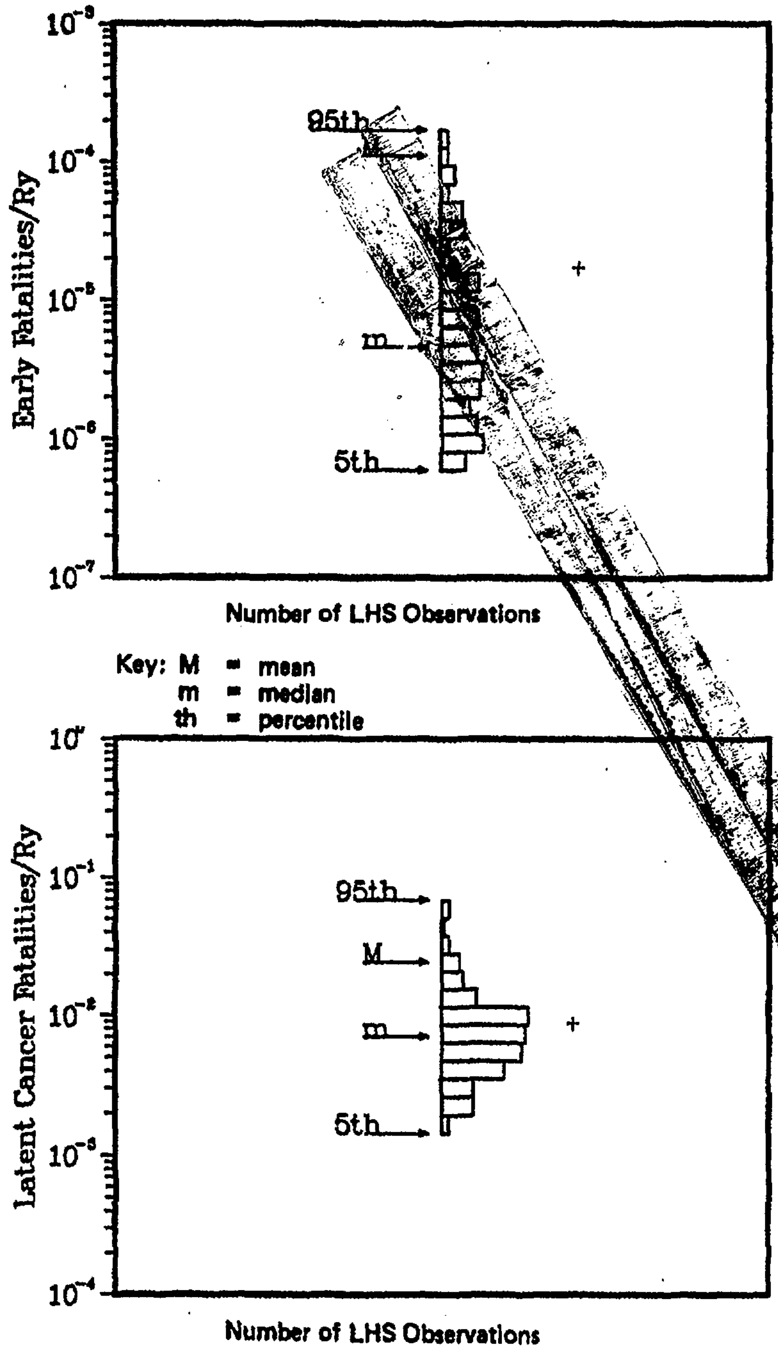
The risk results shown in Figure 7.8 have been analyzed to identify the principal contributors (accident sequences and containment failure modes) to plant risk. These results are presented in Figures 7.11 and 7.12. As may be seen, both for early and latent cancer fatality risks, the dominant plant damage state is loss-of-coolant-accident (LOCA) sequences, which have the highest relative frequency and relatively high release fractions. Zion plant risks are dominated by early containment failure (alpha-mode failure, containment isolation failure, and overpressurization

*As noted in Section 7.2, sensitivity studies have been performed to reflect recent modifications in the Zion plant. The impact on risk is displayed on the figures in this section. More detailed discussion on the sensitivity studies may be found in Section C.15 of Appendix C.



Note: As discussed in Reference 7.6, 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.

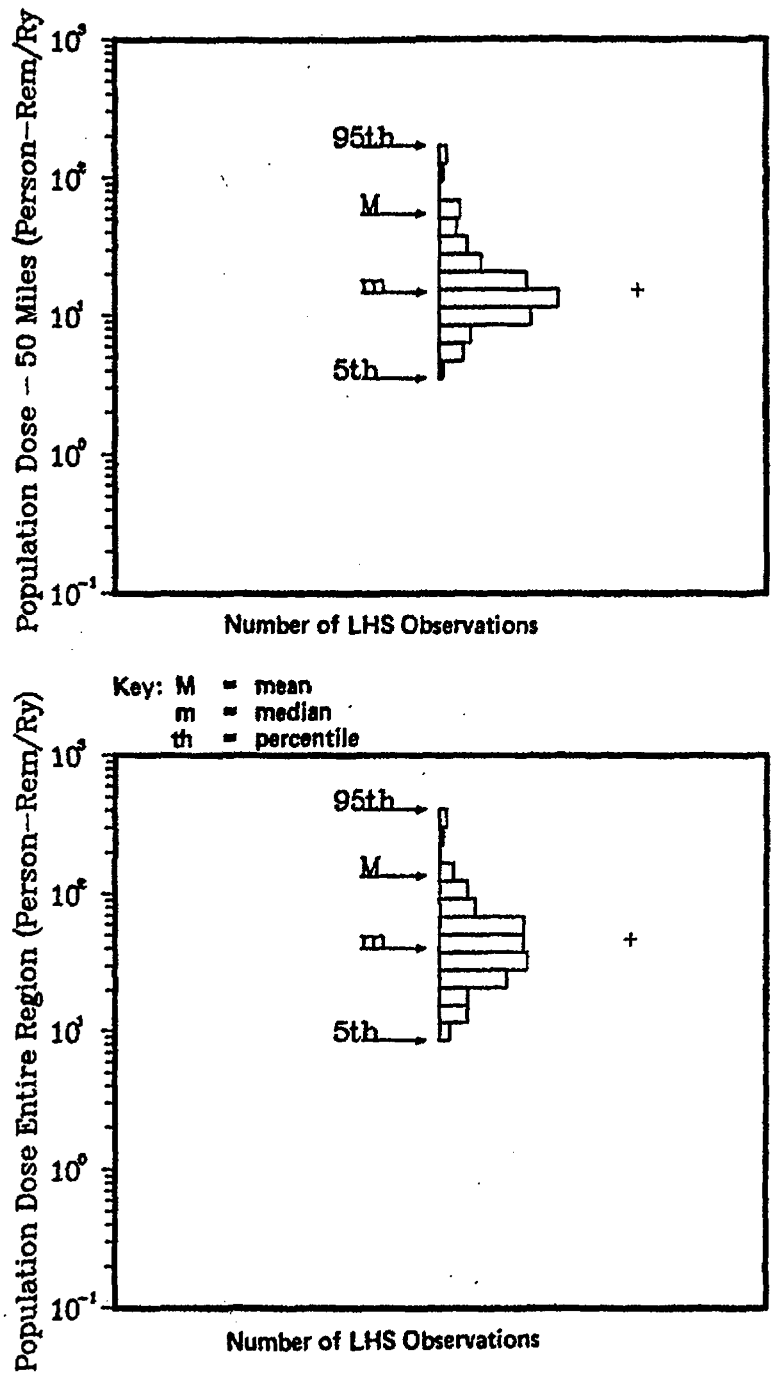
Figure 7.7 Frequency distributions of offsite consequence measures at Zion (internal initiators).



Notes As discussed in Reference 7.6, 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.

"+" shows recalculated mean value based on plant modifications discussed in Section 7.2.1.

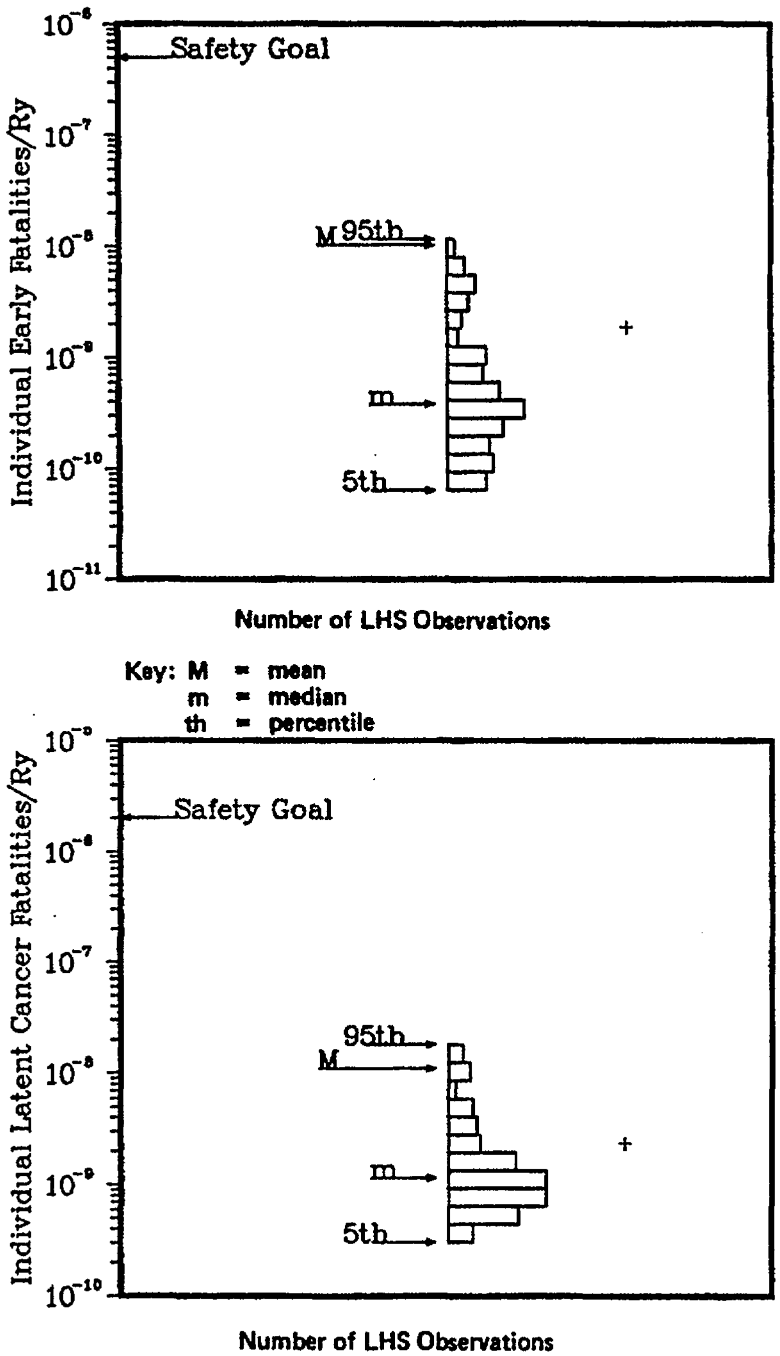
Figure 7.8 Early and latent cancer fatality risks at Zion (internal initiators).



Notes: As discussed in Reference 7.6, 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.

"+"shows recalculated mean value based on plant modifications discussed in Section 7.2.1.

Figure 7.9 Population dose risks at Zion (internal initiators).



Notes: As discussed in Reference 7.6, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of other health effects not studied in the risk analyses.

"+"shows recalculated mean value based on plant modifications discussed in Section 7.2.1.

Figure 7.10 Individual early and latent cancer fatality risks at Zion (internal initiators).

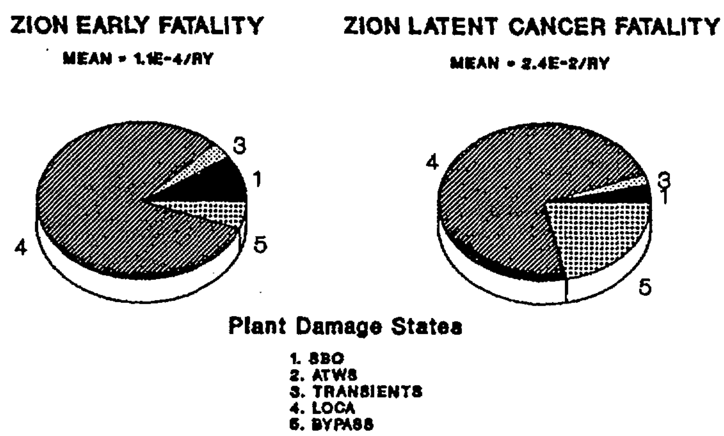


Figure 7.11 Major contributors (plant damage states) to mean early and latent cancer fatality risks at Zion (internal initiators).

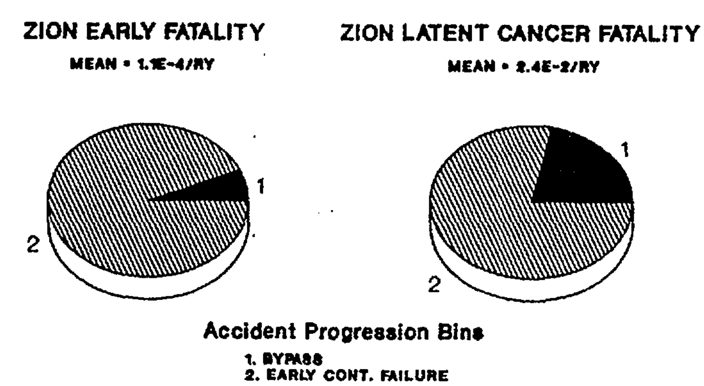


Figure 7.12 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Zion (internal initiators).

failure). This occurs because, although the conditional probability of early failure is low, other failure modes have even lower probabilities.

7.6.2 Important Plant Characteristics (Risk)

- As discussed before, the dominant risk contributor for the Zion plant is early containment failure. The accident progression bin for early containment failure contains several failure modes such as the alpha-mode, containment isolation, and overpressurization failures.
- The containment structure at Zion is robust, with a low probability of failure. This has led to the low risk estimates from the Zion plant. (In comparison with other plants studied in this report, risks from Zion are relatively high; but, in the absolute sense, the risks are very low and well below the NRC safety goals.)

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