ATTACHMENT 1

FirstEnergy's Statement of Material Facts on Which There is No Genuine Issue to be Heard (July 26, 2012)

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket No. 50-346-LR
FIRSTENERGY NUCLEAR OPERATING COMPANY	)	
	)	
(Davis-Besse Nuclear Power Station, Unit 1)	)	July 26, 2012

FIRSTENERGY'S STATEMENT OF MATERIAL FACTS ON WHICH THERE IS NO GENUINE DISPUTE TO BE HEARD

FirstEnergy Nuclear Operating Company ("FirstEnergy") submits this statement of undisputed material facts in support of its Motion for Summary Disposition of Contention 4 (SAMA Analysis Source Terms) (July 26, 2012).

I. REGULATORY AND TECHNICAL BACKGROUND

A. Submittal of the Original and Revised Davis-Besse SAMA Analyses

1. On August 27, 2010, FirstEnergy submitted a license renewal application ("LRA"), requesting that the NRC renew the operating license for Davis-Besse Nuclear Power Station, Unit 1 ("Davis-Besse") for 20 years (*i.e.*, through April 22, 2037). Letter from Barry S. Allen, FirstEnergy, to NRC Document Control Desk, "License Renewal Application and Ohio Coastal Zone Management Program Consistency Certification" (ADAMS Accession No. ML102450572 (package)).

2. As required by 10 C.F.R. § 51.53(c)(3)(ii)(L), FirstEnergy prepared an analysis of severe accident mitigation alternatives ("SAMAs") as part of its LRA. The Davis-Besse SAMA analysis is documented in Section 4.20 and Attachment E of the Environmental Report ("ER"). ER § 4.20 (Severe Accident Mitigation Alternatives) & Attach. E (Severe Accident Mitigation Alternatives Analysis).

3. On July 16, 2012, FirstEnergy submitted to the NRC certain revisions to the SAMA analysis documented in ER Section 4.20 and ER Attachment E. Among other things, the revised

SAMA analysis accounts for FirstEnergy's use of updated Modular Accident Analysis Program ("MAAP") code runs that, consistent with MAAP User's Group recommendations, are based on core inventory radionuclide masses instead of radionuclide activities. Letter from John C. Dominy, Director, Site Maintenance, FirstEnergy, to NRC Document Control Desk, "Correction of Errors in the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4613) Environmental Report Severe Accident Mitigation Alternatives Analysis, and License Renewal Application Amendment No. 29" (July 16, 2012) ("Revised SAMA Analysis Submittal") (Attach. 5).

B. SAMA Analysis Requirements and Guidance

4. SAMAs, by definition, pertain to severe accidents; *i.e.*, accidents in which substantial damage is done to the reactor core, whether or not there are serious offsite consequences. Policy Statement on Severe Reactor Accidents Regarding Future Design and Existing Plants, 50 Fed. Reg. 32,138 (Aug. 8, 1985) (Attach. 11); Joint Declaration of Kevin O'Kula and Grant Teagarden in Support of FirstEnergy's Motion for Summary Disposition of Contention 4 (SAMA Analysis Source Terms) at ¶ 15 (July 26, 2012) ("Joint Decl.") (Attach. 2).

5. NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," Vol. 1, at 5-1 to 5-20 (May 1996) ("GEIS") (Attach. 12), provides an evaluation of severe accident impacts that applies to all U.S. nuclear power plants. Based on the GEIS evaluation of severe accident impacts, 10 C.F.R. Part 51 concludes that the "[t]he probability weighted consequences of atmospheric releases, fallout onto open bodies of water, releases to ground water, and societal and economic impacts from severe accidents are small for all plants." 10 C.F.R. Part 51, Subpart A, App. B, Table B-1 (Postulated Accidents; Severe accidents); Joint Decl. ¶ 15.

6. 10 C.F.R. Part 51 states that if the Staff has not previously considered SAMAs for a license renewal applicant's plant in an EIS or in an environmental assessment, then the applicant must complete an evaluation of alternatives to mitigate severe accidents. 10 C.F.R. § 51.53(c)(3)(ii)(L); *see also* 10 C.F.R. Part 51, Subpart A, App. B, Table B-1; Joint Decl. ¶ 16.

7. SAMA analysis is a site-specific, probability-weighted assessment of the benefits and costs of mitigation alternatives that might be used to reduce the risks (frequencies or consequences or both) of potential nuclear power plant severe accidents. It estimates annual

average impacts for the entire 50-mile radius region surrounding a nuclear power plant. Joint Decl. ¶ 17.

8. The Nuclear Energy Institute ("NEI") has issued a guidance document, NEI 05-01, Revision A, to assist NRC license renewal applicants in preparing SAMA analyses. NEI 05-01, Rev. A, "Severe Accident Mitigation Alternatives (SAMA) Analysis, Guidance Document," at i (Nov. 2005) ("NEI 05-01") (Attach. 14); Joint Decl. ¶ 18.

9. The Staff has approved and recommended the use of NEI 05-01 by license renewal applicants. "Final License Renewal Interim Staff Guidance LR-ISG-2006-03: Staff Guidance for Preparing Severe Accident Mitigation Alternatives Analyses" (Aug. 2007) (Attach. 15); Joint Decl. ¶ 18.

10. NEI 05-01 states: "The purpose of the analysis is to identify SAMA candidates that have the potential to reduce severe accident risk and to determine if implementation of each SAMA candidate is cost-beneficial." NEI 05-01 at 1 (Attach. 15); Joint Decl. ¶ 18.

11. A SAMA analysis identifies potential changes to a nuclear power plant, or its operations, that could reduce the already low risk (frequency and/or the consequence) of a severe accident for which the benefit of implementing the change may outweigh the cost of implementation. Changes to the plant that could reduce the risk of a severe accident include plant modifications or operational changes (*e.g.*, improved procedures, augmented training of control room and plant personnel). NEI 05-01 at 1, 23 (Attach. 15); Joint Decl. ¶ 18.

12. A SAMA analysis, broadly speaking, involves four major sequential steps: (1) using probabilistic risk assessments ("PRAs") and other risk studies to characterize the overall plant-specific severe accident risk by identifying and characterizing the leading contributors to core damage frequency ("CDF") and offsite risk based on a plant-specific risk study; (2) identifying potential plant improvements (*i.e.*, SAMA candidates) that could reduce the risk of a severe accident; (3) quantifying the risk-reduction potential and the implementation cost for each SAMA candidate; and (4) determining whether implementation of the SAMA candidates may be cost-effective. NEI 05-01 at 2 (Attach. 15); Joint Decl. ¶ 19.

13. The SAMA evaluation of a plant is based on the numerical evaluation of severe accident risk impacts in four categories: (1) offsite exposure cost, (2) offsite economic cost, (3)

onsite exposure cost, and (4) onsite economic cost. This methodology for the overall SAMA analysis approach is based on methods found in NRC guidance. NEI 05-01 at 28 (Attach. 15); NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," Rev. 4 (Jan. 1997) (Attach. 16); NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission, Revision 4" (August 2004) (Attach 17); Joint Decl. ¶ 19.

C. Use of Plant-Specific Probabilistic Risk Assessment ("PRA") in SAMA Analyses

14. The basis for a SAMA analysis conducted for a U.S. nuclear power plant is a sequential, three-level probabilistic risk assessment or PRA. All three PRA levels are required to perform a SAMA analysis. Joint Decl. ¶ 21.

15. The Level 1 PRA establishes the plant damage states and frequency of reactor core damage frequency or CDF. Joint Decl. ¶¶ 47-48.

16. The Level 2 PRA determines different accident progressions and a set of radioactive release conditions from the containment that are assigned to similar representative groups (release categories). The Level 2 PRA defines the sequence of events resulting in a radioactive release to the environment. The source term analysis then follows and quantifies the amount of radioactivity released for a given sequence and the frequency of occurrence (*i.e.*, release categories and their respective frequencies). Joint Decl. ¶¶ 47-48.

17. The Level PRA 3 combines the Level 2 PRA results with site-specific parameters (*e.g.*, population distribution, meteorological data, land use data, and economic data) for the Level 3 PRA to calculate offsite public dose and offsite economic consequences of those releases to the environment. Joint Decl. ¶¶ 47-48.

D. Use of the MAAP Code to Develop Source Term Inputs to the MACCS2 Code

18. Various computer codes are used in support of a SAMA analysis. These codes include, among others, the MELCOR Accident Consequence Code System Version 2 ("MACCS2") and the Modular Accident Analysis Progression ("MAAP") codes. Joint Decl. ¶ 20.

19. As part of the Level 3 PRA, MACCS2 calculates the radiological doses, health effects, and economic consequences that result from postulated releases of radioactive materials to the atmosphere. MACCS2 performs these calculations based on plant- and site-specific, regional, and standardized regulatory inputs. NEI 05-01 at 13 (Attach. 15); Joint Decl. ¶ 20.

20. MACCS2 executes three modules (ATMOS, EARLY, and CHRONC) in sequence to calculate consequence values necessary for a SAMA analysis, and models atmospheric transport and dispersion and subsequent deposition in a radial-polar grid (*i.e.*, 16 compass sectors over a 50-mile radius). NUREG/CR-6613, "Code Manual for MACCS2: User's Guide," Vol. 1 at 2-1 to 2-3 (May 1998) (Attach. 19); Joint Decl. ¶ 23.

21. ATMOS, in particular, performs calculations pertaining to atmospheric transport, dispersion, and deposition of radioactive material, and to radioactive decay of that material both before and after its release into the atmosphere. NUREG/CR-6613 at 2-2 (Attach. 19); Joint Decl. ¶ 23.

22. The ATMOS input parameters include, among other things, plant-specific source term information, including the core inventory (*i.e.*, the amount of each radionuclide present in the reactor core at the time of accident initiation), and the physical, chemical, and radiological composition of an atmospheric release. NUREG/CR-6613 at 5-23 to 5-28 (Attach. 19); Joint Decl. ¶ 24.

23. The source term is the amount and isotopic composition of material released (or postulated to be released) from the core of a nuclear power reactor during an accident. NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Vol. 1, 2-3 tbl. 2.1 (Dec. 1990) (Attach. 10); Joint Decl. ¶ 24.

24. The source term may refer to radionuclide groups in the reactor core inventory at the start of an accident that are released to the containment (*i.e.*, the containment source term) or that are released to the environment (*i.e.*, the environmental source term). Joint Decl. ¶ 25.

25. An environmental source term describes the physical, chemical, and radiological composition of an atmospheric release. The environmental source term description includes: (1) the quantity of each important radionuclide released into the atmosphere, (2) the initial time of the release relative to the start of the accident, (3) the duration of the release, (4) the elevation of the release, (5) the sensible heat released, and (6) the particle size of the released material. Joint Decl. ¶¶ 24-25.

26. The release fraction, *i.e.*, the fraction of the total activity of the fission products released to the environment during the accident, is one component of the source term. It defines

the portion of the radionuclide inventory, by radionuclide group, in the reactor core at the start of an accident that is ultimately released to the environment. Joint Decl. ¶ 26.

27. Source terms depend on how rapidly the accident progresses, the path by which the radionuclides escape from the reactor into containment, the path through containment (or possibly bypassing containment altogether), and the effectiveness of both passive and active safety features that are intended to mitigate releases. Joint Decl. ¶ 27.

28. The evaluation of source terms for a SAMA analysis requires use of a detailed analytical model that includes a multitude of physical process sub-models that account for, among other things, the timing and performance of both passive and active plant safety features, and human (*i.e.*, operator) actions affecting accident progression and containment conditions. Joint Decl. ¶ 27.

29. Source terms commonly are estimated in the U.S. using one of two computer codes: the MAAP code and the Methods for Estimation of Leakages and Consequences of Releases ("MELCOR") code. Joint Decl. ¶ 28.

30. FirstEnergy used MAAP4 (Version 4.0.6) in support of the Davis-Besse SAMA analysis. Joint Decl. ¶ 28.

31. MAAP simulates the dominant thermal-hydraulic and fission product phenomena in both the primary and containment systems of pressurized water reactors ("PWRs") and boiling water reactors ("BWRs"). MAAP thus evaluates a broad spectrum of phenomena, including steam formation; core heat-up; cladding oxidation and hydrogen evolution; vessel failure; corium-concrete interactions; ignition of combustible gases; fluid entrainment by high-velocity gases; and fission-product release, transport, and deposition. MAAP4 also addresses important engineered safety systems and allows a user to model operator interventions. EPRI Report 1020236, "MAAP4 Applications Guidance: Desktop Reference for Using MAAP4 Software, Revision 2" at 2-2 to 2-6 (2010) ("MAAP4 Applications Guidance") (Attach. 20); Fauske & Associates, LLC, *MAAP (Modular Accident Analysis Program)* (Attach. 21); Joint Decl. ¶ 29.

E. Use of Plant-Specific PRAs and MAAP4 Code in the Davis-Besse SAMA Analysis

32. The progression from the failure of individual plant components to the determination of accident frequencies, accident progressions, and offsite consequences involves

plant- and site-specific phenomena and can be separated into the three PRA levels. For its SAMA analysis, FirstEnergy used the results from updated Davis-Besse Level 1 and Level 2 PRA models as input to a Level 3 PRA model developed specifically to support the consequence quantification needed for the Davis-Besse SAMA analysis. Joint Decl. ¶¶ 47-48.

33. The Level 1 PRA included initiating event and core damage sequence analyses and yielded a set of plant damage states and associated frequencies. Joint Decl. ¶ 48.

34. The Level 2 PRA used containment event tree ("CET") and deterministic source term modeling to provide a set of 34 release categories, each of which has a characteristic frequency and unique timing and fission product magnitude characteristics, depicting the release to the environment. The release categories are defined in terms of similar properties, each with a frequency-weighted mean source term. Joint Decl. ¶ 48.

35. The Level 2 PRA-defined release categories were characterized using the MAAP4 code. The MAAP4 calculations provided a deterministic analysis of the plant under postulated severe accident conditions for a variety of initiating events, and included the influence of operator actions and safety system actuation on accident sequence progression. The MAAP4 calculations predicted the integrated response of the reactor core, primary system, steam generators, and primary containment building. Results included the time of core damage and reactor vessel failure to support Level 1 PRA success criteria, as well as containment response and fission product source term characterization to support the Level 2 and Level 3 assessments. Joint Decl. ¶ 51.

36. Six MACCS2 input parameters came from the output of the Davis-Besse MAAP4 runs: (1) the time after accident initiation that the offsite alarm is initiated (OALARM), (2) the heat content of release segment (PLHEAT), (3) the height of the plume segment at release (PLHITE), (4) the duration of release (PLUDUR), (5) the time of release for each plume (PDELAY), and (6) the release fraction for each radioisotopic group (RELFRC). Joint Decl. ¶ 52.

37. The core inventory for the Davis-Besse Level 3 PRA was obtained from plant-specific calculations performed using the ORIGEN-2 code. For conservatism, the Davis-Besse core inventory was evaluated at the 24-month end-of-cycle for all 177 fuel assemblies. This assumption is conservative because at the end-of-cycle, the radionuclide quantities in the core would be at their peak levels for the 24-month cycle. In total, 58 radionuclides were evaluated in

the MACCS2 reactor core inventory for Davis-Besse and are represented in nine fission product groups. Joint Decl. ¶ 52.

38. The release category frequencies and characterizations developed using Level 2 PRA information and MAAP4 were used as inputs to the Level 3 PRA. The Level 2 PRA results were then combined with Davis-Besse site-specific parameters (*e.g.*, population distribution, meteorological data, land use data, and economic data) for the Level 3 PRA to estimate the Davis-Besse Plant offsite population dose risk (in units of person-rem/year) and offsite economic cost risk (in units of dollars/year), the key risk metrics in a SAMA analysis. Joint Decl. ¶¶ 47-48.

II. ISSUES RAISED IN INTERVENORS' CONTENTION 4

39. Contention 4 alleges that FirstEnergy's SAMA analysis "underestimates the true cost of a severe accident at Davis-Besse." Beyond Nuclear, Citizens Environment Alliance of Southwestern Ontario, Don't Waste Michigan, and the Green Party of Ohio Request for Public Hearing and Petition for Leave to Intervene at 100, 104, 108 (Dec. 27, 2010) ("Petition") (Errata filed Jan. 5, 2011); *FirstEnergy Nuclear Operating Co.* (Davis-Besse Nuclear Power Station, Unit 1), LBP-11-13, slip op. at 50-54, 64 (Apr. 26, 2011), *aff'd in part and rev'd in part*, *FirstEnergy Nuclear Operating Co.* (Davis-Besse Nuclear Power Station, Unit 1), CLI-12-08, (Mar. 27, 2012).

40. Contention 4 further alleges that FirstEnergy has minimized the potential amount of radioactive material released in a severe accident by using MAAP-derived source terms that are smaller for key radionuclides than the release fractions specified in NRC guidance. Petition at 108. Intervenors make three principal claims in support of their contention (which, for clarity and ease of reference, FirstEnergy refers to as Bases 1, 2 and 3):

1. The MAAP code "has not been validated by the NRC." *Id.* (Basis 1)
2. The radionuclide release fractions generated by MAAP "are consistently smaller for key radionuclides than the release fractions specified in NUREG-1465" and result in "anomalously low" accident consequences. *Id.* at 108, 112, 114 (Basis 2)
3. It previously has been observed that MAAP generates lower release fractions than those derived and used by NRC in other severe accident studies. *Id*. at 113. (Basis 3)

III. UNDISPUTED FACTS SHOWING LACK OF GENUINE MATERIAL DISPUTE

A. Validation of the MAAP Code (Basis 1 of Contention 4)

41. MAAP was originally developed for the Industry Degraded Core Rulemaking ("IDCOR") program in the early 1980s by Fauske & Associates, LLC (formerly Fauske & Associates, Inc.). At the completion of IDCOR, ownership of MAAP was transferred to the Electric Power Research Institute ("EPRI"), which was charged with maintaining and improving the code. Fauske & Associates, LLC is the current maintenance contractor for the MAAP code. MAAP4 Applications Guidance at 2-2 (Attach. 20); Joint Decl. ¶¶ 31, 33;

42. Starting in the late 1980s, the MAAP3B version became widely used, first in the United States and then worldwide, to support success criteria determination, human action timing evaluations, and Level 2 analyses for Individual Plant Examinations ("IPEs"). MAAP4 Applications Guidance at 2-2 (Attach. 20); Joint Decl. ¶ 31.

43. MAAP3B updated to MAAP4 in the mid-1990s to expand its modeling capabilities. MAAP4 incorporates updated physical models for core melt, reactor vessel lower head response, and containment response that provide improved mechanistic modeling of severe accident phenomena. MAAP4 Applications Guidance at 2-2 (Attach. 20); Joint Decl. ¶ 31.

44. Several organizations, including EPRI and the DOE, sponsored the development of MAAP4. As part of the development process, a committee of independent experts reviewed MAAP4 to ensure that it is state-of-the-art and applicable for accident management evaluations. Further, the new software was subjected to review by a Design Review Committee, comprised of senior members of the nuclear safety community. MAAP4 Applications Guidance at 2-2 (Attach. 20); Joint Decl. ¶ 31.

45. MAAP and its successor versions, including MAAP4, were developed in accordance 10 C.F.R. Part 50, Appendix B and International Organization for Standardization ("ISO") 9001 quality assurance requirements. MAAP4 Applications Guidance at 2-2 (Attach. 20); Joint Decl. ¶ 33.

46. EPRI has identified the MAAP code (versions 4.0.5 and later) as a "consensus model" suitable for use in evaluation of PRA success criteria. EPRI Report 1013492,

"Probabilistic Risk Assessment Compendium of Candidate Consensus Models" at 2-3 (2006) (Attach. 23); Joint Decl. ¶ 33.

47. MAAP4 has been benchmarked against numerous severe accident studies and the Three Mile Island Unit-2 ("TMI-2") core melt accident. The benchmarking of MAAP is documented in Section 7 (MAAP Benchmarks) and Appendix F (Summaries of MAAP Benchmarks) of EPRI's MAAP4 Applications Guidance and also in the Nuclear Energy Agency's Committee on the Safety of Nuclear Installations report "Recent Developments in Level 2 PSA and Severe Accident Management." Committee on the Safety of Nuclear Installations, Nuclear Energy Agency, Organization for Economic Co-operation and Development, NEA/CSNI/R(2007)16, at 36 (Nov. 2007) (Attach. 24); Joint Decl. ¶ 34.

48. EPRI licenses MAAP4 to a wide array of entities, such as utilities, vendors, and research organizations, including universities. The majority of MAAP4 users are members of the MAAP Users Group ("MUG"). The MUG provides direction and funding for code maintenance, enhancements, and benchmarking; facilitates information transfer through biannual meetings and the issuance of various communications on code problems and best practices; and supports industry and regulatory acceptance. MAAP4 Applications Guidance at 2-2 (Attach. 20); Joint Decl. ¶ 32.

49. In general, a computer code in itself is not validated by the NRC, but its use for specific applications may be found acceptable for estimating certain phenomena within certain defined regimes. For example, a computer code may be used to predict the coupled thermal-hydraulic fission product transport response of reactor systems to severe accident events. If inputs and assumptions are appropriate for the computer model, and sources of uncertainty are understood, then the results of that code may be accepted by a reviewer or regulator for purposes of the application. Joint Decl. ¶ 30; Letter from Gary M. Holahan, Director, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation, U.S. N.R.C., to Theodore U. Marston, Vice-President & Chief Nuclear Officer, EPRI at 1 (Dec. 4, 2001) (Attach. 22).

50. The MAAP code has been used by nuclear plant licensees and other entities to predict the responses of nuclear power plants during postulated severe accidents and is the most commonly used code in the U.S. for such purposes. Joint Decl. ¶ 35; Kenneth D. Kok, Ed., *Nuclear Engineering Handbook* at 539 (2009) (Attach. 25).

51. The use of MAAP and its successor versions in IPEs and subsequent PRA applications, including those related to advanced reactor standard design certification applications, has been accepted by the NRC Staff. *See*, *e.g.*, NUREG-1503, "Final Safety Evaluation Report Related to Certification of the ABWR Reactor Design," Vol. 1 at 19-53 to 19-55 (July 1994) (Attach. 47); NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," Vol. 1 at 19-61 (Sept. 2004) (Attach. 48); Joint Decl. ¶ 35.

52. Numerous NRC license renewal applicants have used the MAAP code to support NRC-approved SAMA analyses, including very recent recipients of renewed operating licenses. *See*, *e.g.*, NUREG-1437, Supp. 47, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Columbia Generating Station – Final Report, Vol. 2, App. F at F-2, F-6 to F-7, F-27 (Apr. 2012) (Attach. 26); NUREG-1437, Supp. 45, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Regarding Hope Creek Generating Station and Salem Nuclear Generating Station, Units 1 and 2, Vol. 2, App. G at G-4, G-6, G-15 to G-16 (Mar. 2011) (Attach. 27); Joint Decl. ¶ 36.

B. <u>Differences in MAAP4-Generated and NUREG-1465 Source Terms (Basis 2)</u>

53. The reactor accident source term generally serves two purposes in the U.S. nuclear regulatory process. The first purpose is for licensing, safety analysis, and regulatory compliance, particularly in meeting 10 C.F.R. Part 100 siting requirements. For this purpose, a source term representing the release of radioactive materials into the reactor containment is used to assess the adequacy of reactor containments and engineered safety systems, as well as the environmental qualification of equipment inside the containment that must function following a design-basis accident. This source term also is used to show that dose criteria at the exclusion area boundary are met by assuming the maximum allowable design leak rate from the containment. NUREG-1465 source terms are applicable for the first purpose described above. 10 C.F.R. § 50.34(a)(1)(ii)(D) & 10 C.F.R. § 100.11; Joint Decl. ¶ 38.

54. The second purpose for which a reactor accident source term is developed is to simulate a release of radioactive material to the environment (*i.e.*, outside containment) following a hypothetical reactor accident. This second source term may be used as input to radionuclide dispersal and accident consequence models (*e.g.*, MACCS2) that are used for Level 3 PRA and SAMA evaluations. Joint Decl. ¶ 39.

55. NUREG-1465 states that it was developed "to define a revised accident source term for regulatory application for future LWRs" and "to provide a postulated fission product source term released into containment that is based on current understanding of LWR accidents and fission product behavior." NUREG-1465, Accident Source Terms for Light-Water Nuclear Power Plants, at vii, 3 (Feb. 1995) (Attach. 8); Joint Decl. ¶ 41.

56. NUREG-1465 provides generic, default source terms, whereas PRA and SAMA analyses are intended to be best-estimate engineering evaluations that seek to maximize the use of plant-specific data. Joint Decl. ¶ 45.

57. NUREG-1465 assumes a "release resulting from 'substantial meltdown' of the core into the containment . . . [and assumes] that the containment remains intact but leaks at its maximum allowable leak rate." NUREG-1465 at vii, 1, 3 (Attach. 8); Joint Decl. ¶ 42.

58. NUREG-1465 discusses in-containment fission product removal mechanisms, such as engineered safety features ("ESFs"), but does not provide numerical estimates of source terms that account for the effects of such mechanisms (*e.g.*, containment sprays, aerosol deposition). NUREG-1465 directs the reader to use appropriate methodologies in crediting fission product removal or reduction within containment. NUREG-1465 at 17-21 (Attach. 8); Joint Decl. ¶ 44.

59. The NUREG-1465 source term solely represents radionuclides released into the containment. It does not specify the source term released from containment into the environment following a severe accident, and it does not take into account the reductions of the source term that would occur in those circumstances. Joint Decl. ¶¶ 42-43.

60. The MAAP code produces results that are different from, and generally smaller than, the release fractions specified in NUREG-1465, because MAAP models the release of radionuclides from the containment into the environment following a postulated severe accident. MAAP models and credits fission product removal mechanisms such as containment ESFs (*e.g.*, containment air coolers, containment spray) and natural depletion processes (*e.g.*, aerosol deposition and containment holdup). Joint Decl. ¶¶ 43-44.

C. Differences in MAAP4-Generated Source Terms and Source Terms Discussed in Other Historical Studies Cited by Intervenors (Basis 3)

61. Contention 4 cites two documents containing historical comparisons between release fractions developed using earlier versions of the MAAP code and release fractions developed using other codes. The first document is a 1987 draft of the NUREG-1150 severe accident risk study that, in examining accident risk at Zion Nuclear Station, stated that "the MAAP estimates for environmental release fractions were significantly smaller" than those obtained with the Source Term Code Package[1] ("STCP") computer code (the primary code used in the NUREG-1150 study). Petition at 114 (citing Office of Nuclear Regulatory Research, "Draft for Comment, Reactor Risk Reference Document," NUREG-1150, Vol. 1, at 5-14 (Feb. 1987) (Attach. 9)). This statement does not appear in the final December 1990 version of NUREG-1150 (Attach. 10). Joint Decl. ¶¶ 58, 59.

62. The second is a 2002 Brookhaven National Laboratory ("BNL") report reviewing combustible gas control availability at ice condenser and Mark III containment plants. The BNL study compared the Level 2 portion of the PRA results for the Catawba plant (obtained using the MAAP code) with a "typical NUREG-1150 release" for the Sequoyah plant (obtained using the STCP and MELCOR codes). John R. Lehner et al., Brookhaven National Laboratory, "Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice Condenser and Mark III Containment Plants, Final Letter Report" at 17 (Dec. 2002) ("BNL report") (Attach. 34); Joint Decl. ¶ 58.

63. The BNL report states that the "NUREG-1150 release fractions for the important radionuclides are about a factor of 4 higher than the ones" in the Catawba PRA, and that the "differences in the release fractions . . . are primarily attributable to the use of the different codes in the two analyses." BNL report at 17 (Attach. 34); Joint Decl. ¶ 58.

64. Severe accident source term estimates depend on many plant-specific design features, operational practices, and the technical accuracy provided by computer code models used for source term quantification Joint Decl. ¶ 59.

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[1] As noted on the Sandia MELCOR website (http://melcor.sandia.gov/), the STCP is the predecessor to MELCOR: "MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light-water reactor nuclear power plants. MELCOR is being developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission as a second-generation plant risk assessment tool and the successor to the Source Term Code package."

65. The NUREG-1150 study (issued as a final report in 1990) was completed over 20 years ago and involved an assessment of the risks from severe accidents at five commercial nuclear power plants in the United States. Davis-Besse was not one of those five plants. Joint Decl. ¶ 59; NUREG-1150 (Attach. 10).

66. The IDCOR (MAAP) to NUREG-1150 (STCP) comparison of Zion results cited by Intervenors was only one of four sets of plant results compared in the February 1987 draft of NUREG-1150. In addition, after extensive peer review of, and public comment on, the February 1987 draft, NUREG-1150, Volume 1, was issued as a second draft in 1989, before being published as a final report in December 1990. The report and its underlying technical analyses were substantially modified in two rounds of review before the report's final publication in December 1990. One of the changes included deleting the specific discussion comparing the MAAP and STCP results for Zion, such that the comparison cited by Intervenors in Contention 4 was not incorporated into the final December 1990 version of NUREG-1150. Joint Decl. ¶ 59; Draft NUREG-1150, Vol. 1 at 5-14 (Attach. 9); NUREG-1150, Vol. 1 (Attach. 10).

67. The final NUREG-1150 report states that the thermal-hydraulic model in the STCP "uses simplified models and assumptions for the treatment of some of the very complex steps in the core degradation process, such as fuel slumping into the lower plenum of a reactor vessel." NUREG-1150, Vol. 3, App. D at D-17 (Attach. 10); Joint Decl. ¶ 60. More realistic models such as MELCOR and MAAP were used to adjust the thermal-hydraulic estimates affecting core degradation, ultimately leading to differences in the estimated source term. NUREG-1150, Vol. 3, App. D at D-17 (Attach. 10); Joint Decl. ¶ 60.

68. The BNL report's comparison between the Catawba Level 2 PRA release fractions and the NUREG-1150 Sequoyah release fractions represents a difference of more than ten years in terms of severe accident modeling (~2002 versus ~1990). The comparison of MAAP-based source terms with those estimated over ten years earlier with STCP (a simpler code) and an earlier version of MELCOR—and for different plants—is expected to show differences. Joint Decl. ¶ 63.

69. Also, the BNL report comparison uses a release category that represents an early containment failure in which the Catawba source term is based on an "early containment failure without ex-vessel release" assumption that may not have been applied in the Sequoyah source term. *See* Memorandum from Asimios Malliakos, Probabilistic Risk Analysis Branch, Division of

Risk Analysis and Applications, Office of Nuclear Regulatory Research, to Marc A. Cunningham, Chief, Probabilistic Risk Analysis Branch, Division of Risk Analysis and Applications, Office of Nuclear Regulatory Research, "Telecommunication with Duke Energy Corporation in Support of Generic Safety Issue (GSI) 189, 'Susceptibility of Ice Condenser and BWR Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident,'" Attach. 1, 3 (Oct. 8, 2002) (Attach. 38); Joint Decl. ¶ 63.

70. Since the issuance of NUREG-1150, better understanding of heat transfer and removal from the reactor pressure vessel during severe accident sequences, improved insights on iodine, cesium, and other fission product groups' chemistry from contemporary research, and modeling improvements suggest that the early containment failure releases would be smaller than previously estimated. Joint Decl. ¶ 64.

71. In its 2002 Supplemental Environmental Impact Statement for Catawba license renewal, the NRC Staff compared similar sequences between the two studies—NUREG-1150 and Revision 2b of the Catawba PRA,[2] which included the plant's IPE models—and concluded there was "reasonable agreement" for the closest corresponding release scenarios. NUREG-1437, Supp. 9, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Regarding Catawba Nuclear Station, Units 1 and 2 – Final Report at 5-9 to 5-10 (Dec. 2002) (Attach. 37); Joint Decl. ¶ 62.

72. The state of the art for source term analysis has significantly improved since the NUREG-1150 study was performed in the 1980s. For example, in 2006, the NRC initiated the State-of-the-Art Reactor Consequence Analyses ("SOARCA") project to develop revised best estimates of the offsite radiological health effect consequences of severe reactor accidents. The project's principal objective was to develop updated and more realistic severe accident analyses by including significant plant changes and reactor safety research updates not reflected in earlier NRC assessments such as WASH-1400, the 1982 Siting Study, and NUREG-1150. SOARCA included consideration of plant system improvements, improvements in training and emergency procedures, offsite emergency response, and security-related improvements, as well as plant changes such as power uprates and lengthened operating times. Joint Decl. ¶ 65; NUREG-1935,

[2] Letter from Gary R. Peterson, Vice President, Duke Energy Corporation, to Document Control Desk, U.S. N.R.C., Attach. 1, "Catawba PRA Revision 2b Summary Results" (Apr. 18, 2001).

"State-of-the-Art Reactor Consequence Analyses (SOARCA) Report, Draft Report for Comment" (Jan. 2012) ("Draft NUREG-1935") (Attach. 39).

73. The SOARCA analyzed two plants that are typical of the two U.S. commercial reactor types, *i.e.*, a BWR plant, the Peach Bottom Atomic Power Station in Pennsylvania, and a PWR plant, Surry Power Station in Virginia. These two plants also took part in earlier accident analyses performed by the NRC, including the seminal WASH-1400 PRA study (1975), the Sandia Siting Study (1982), and the NUREG-1150 (1990) study. The Staff analyzed one plant unit at each site. Joint Decl. ¶ 66; Draft NUREG-1935 (Attach. 39).

74. The SOARCA project used computer-modeling techniques to understand how a reactor might behave under severe accident conditions and how a release of radioactive material from the plant might affect the public. Specifically, it used MELCOR to model the severe accident scenarios within the plant, and MACCS2 to model the offsite health effect consequences of any atmospheric releases of radioactive material. Joint Decl. ¶ 68; Draft NUREG-1935 (Attach. 39).

75. In January 2012, the NRC published the results of its SOARCA assessment, including plant-specific reports for Peach Bottom and Surry. Among the findings, the NRC found that, in addition to delayed radiological releases, the magnitude of the radionuclide release, especially with respect to the key radioisotopic (iodine and cesium) groups, is much smaller than estimated in prior studies. Joint Decl. ¶ 69; Draft NUREG-1935 (Attach. 39); NUREG/CR-7110, "State-of-the-Art Reactor Consequence Analyses Project: Volume 1: Peach Bottom Integrated Analysis" (Jan. 2012) (Attach. 40); NUREG/CR-7110, "State-of-the-Art Reactor Consequence Analyses Project: Volume 2: Surry Integrated Analysis" (Jan. 2012) (Attach. 42); NUREG/BR-0359, "Modeling Potential Reactor Accident Consequences" (Jan. 2012) (Attach. 43).

76. Both MAAP and MELCOR were used soon after of the March 2011 Fukushima Dai-ichi nuclear power plant accident in Japan. Tokyo Electric Power Company, the operating utility for the six-unit station, has used MAAP to inform its understanding of the accident progression in Units 1-3 during the earthquake and subsequent tsunami event in March 2011. International Atomic Energy Agency, *IAEA International Fact Finding Expert Mission of the Fukushima Dai-ichi NPP Accident Following the Great East Japan Earthquake and Tsunami* at

33-35 (June 2011) (Attach. 45). Sandia applied MELCOR in modeling the Station Blackout sequence for the NRC in support of the Japanese Government. Joint Decl. ¶ 72.

Executed in Accord with 10 C.F.R. § 2.304(d)

Signed (electronically) by Martin J. O'Neill

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COUNSEL FOR FIRSTENERGY

Dated in Washington, DC
this 26th day of July 2012

ATTACHMENT 2

Joint Declaration of Kevin O'Kula and Grant Teagarden in Support of FirstEnergy's Motion for Summary Disposition of Contention 4 (SAMA Analysis Source Terms) (July 26, 2012)

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket No. 50-346-LR
FIRSTENERGY NUCLEAR OPERATING COMPANY	)	
	)	
(Davis-Besse Nuclear Power Station, Unit 1)	)	
	)	July 26, 2012

JOINT DECLARATION OF KEVIN O'KULA AND GRANT TEAGARDEN IN SUPPORT OF FIRSTENERGY'S MOTION FOR SUMMARY DISPOSITION OF CONTENTION 4 (SAMA ANALYSIS SOURCE TERMS)

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JOINT DECLARATION OF KEVIN O'KULA AND GRANT TEAGARDEN IN SUPPORT OF FIRSTENERGY'S MOTION FOR SUMMARY DISPOSITION OF INTERVENORS' CONTENTION 4 (SAMA ANALYSIS SOURCE TERMS)

Kevin R. O'Kula and Grant A. Teagarden state as follows under penalties of perjury:

I. PROFESSIONAL QUALIFICATIONS

A. Dr. Kevin R. O'Kula ("KRO")

1. (KRO) My name is Kevin R. O'Kula. I am an Advisory Engineer with URS Safety Management Solutions ("URS") LLC, in Aiken, South Carolina. I am a consultant to FirstEnergy Nuclear Operating Company ("FirstEnergy") on source term and severe accident mitigation alternatives ("SAMA") analysis issues.

2. (KRO) My professional qualifications are provided in Attachment 3. In brief, I have over 29 years of experience as a technical professional and manager in the areas of safety analysis methods and guidance development, computer code validation and verification, probabilistic risk assessment ("PRA"), deterministic and probabilistic accident and consequence analysis applications for reactor and non-reactor nuclear facilities, source term evaluation, risk management, software quality assurance ("SQA"), and shielding. I obtained my B.S. in Applied

and Engineering Physics from Cornell University in 1975, and my M.S. and Ph.D. in Nuclear Engineering from the University of Wisconsin in 1977 and 1984, respectively.

3. (KRO) In addition, I have over 20 years of experience using and applying the MELCOR Accident Consequence Code System 2 ("MACCS2") computer code and its predecessor, MACCS. I co-chaired a U.S. Department of Energy ("DOE") Accident Phenomenology and Consequence evaluation program in the 1990s that evaluated applicable computer models for radiological dispersion and consequence analysis. More recently, I was a technical peer reviewer of the Sandia National Laboratories ("Sandia") and NRC State-of-the-Art Reactor Consequence Analyses ("SOARCA") Project that reviewed updated and more realistic evaluations of severe accident progression in U.S. nuclear power plants. By virtue of my training and experience, I also am familiar with the Modular Accident Analysis Program ("MAAP") and similar codes and the manner in which they are typically used to support severe accident analyses, PRAs, and SAMA analyses.

4. (KRO) I have taught MACCS2 training courses for DOE and its contractors at Lawrence Livermore National Laboratory, Los Alamos National Laboratory, Idaho National Laboratory, Oak Ridge, the Waste Isolation Pilot Plant, and the DOE Safety Basis Academy. In addition, I was the lead author of a DOE guidance document on the use of MACCS and MACCS2 for DOE safety analysis applications, and managed overall completion of equivalent reports for DOE on MELCOR (similar in function to MAAP) and GENII (similar in function to MACCS2). As part of the SOARCA Project Peer Review Committee, I provided critical review and comment to Sandia and the NRC on the use of integrated modeling of accident progression and offsite consequences from postulated severe accidents using both improved computational analysis tools and more accurate inputs and assumptions reflecting current-day plant operations and accident management/response planning.

5. (KRO) I am providing this Joint Declaration in support of the "Applicant's Motion for Summary Disposition of the Petitioners' Contention 4 (SAMA Analysis Source Terms)" in the above-captioned proceeding. I understand that, in August 2010, FirstEnergy submitted a license renewal application ("LRA") to the U.S. Nuclear Regulatory Commission ("NRC") seeking to renew its operating license for the Davis-Besse Nuclear Power Station, Unit No. 1 ("Davis-Besse") for another 20 years.[1] The SAMA analysis is described in Section 4.20 and Attachment E of the Environmental Report ("ER") (Appendix E to the LRA).[2]

6. (KRO) I have thoroughly reviewed the various inputs and assumptions used in FirstEnergy's SAMA analysis, as submitted in August 2010 and revised in July 2012,[3] to calculate offsite consequences associated with a postulated severe accident at Davis-Besse, including relevant supporting technical documentation for the SAMA analysis prepared by FirstEnergy vendors. I also have reviewed the SAMA analysis revisions and clarifications that FirstEnergy provided in response to NRC Staff requests for additional information ("RAIs") in June and September 2011.[4] I thus have personal knowledge of the modeling methods, inputs, and assumptions used in the Davis-Besse SAMA analysis, as described in the Davis-Besse ER and other related documentation.

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1 *See generally* Letter from Barry S. Allen, Vice President-Nuclear, FirstEnergy, to Document Control Desk, U.S. N.R.C., "License Renewal Application and Ohio Coastal Management Program Consistency Certification" (Aug. 27, 2010) (ADAMS Accession No. ML102450565).

2 *See* ER § 4.20 (Severe Accident Mitigation Alternatives) & Attach. E (Severe Accident Mitigation Alternatives Analysis).

3 Letter from John C. Dominy, Director, Site Maintenance, FirstEnergy, to Document Control Desk, U.S. N.R.C., "Correction of Errors in the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4613) Environmental Report Severe Accident Mitigation Alternatives Analysis, and License Renewal Application Amendment No. 29 (July 16, 2012) ("Revised SAMA Analysis Submittal") (Attach. 5).

4 *See* Letter from Kendall W. Byrd, Director, Site Performance Improvement, FirstEnergy, to Document Control Desk, U.S. N.R.C., "Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4613), Environmental Report Severe Accident Mitigation Alternatives Analysis, and License Renewal Application Amendment No. 10 (June 24, 2011) (Attach. 6); Letter from Barry S. Allen, FirstEnergy, to Document Control Desk, U.S. N.R.C., "Reply to Supplemental Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4613) Environmental Report Severe Accident Mitigation Alternatives Analysis" (Sept. 1, 2011) (Attach. 7).

7. (KRO) In preparing this Joint Declaration, I also reviewed relevant pleadings of the parties and Orders issued by the Atomic Safety and Licensing Board ("Board") and the Commission, applicable NRC regulations and guidance documents, and relevant technical reports and studies.

B. Mr. Grant A. Teagarden ("GAT")

8. (GAT) My name is Grant A. Teagarden. I am Manager for Consequence Analysis for ERIN Engineering and Research, Inc., in Campbell, California. I am acting as a consultant to FirstEnergy on source term and SAMA analysis issues.

9. (GAT) My professional qualifications are provided in Attachment 4. Briefly summarized, I have 14 years of experience in the nuclear field, including 10 years as a manager and technical professional in the areas of PRA, source term analysis, consequence analysis, and nuclear power plant security risk assessment. I obtained a B.S. degree in Mechanical Engineering from University of Miami in 1990 and completed the Bettis Reactor Engineering School at the Bettis Atomic Power Laboratory as part of my training in the U.S. Navy nuclear program. I am a member of the American Nuclear Society ("ANS") and serve as the Vice Chair for the writing committee for ANSI/ANS-58.25, *Standard for Radiological Accident Offsite Consequence Analysis (Level 3 PRA) to Support Nuclear Installation Applications*.

10. (GAT) I am experienced with Level 2 PRA (*e.g.*, severe accident analysis) and the use of the thermal-hydraulic and fission product MAAP code to model severe accident phenomenology. My experience includes using MAAP to develop Level 2 PRA models for commercial nuclear power plants in the United States. I also have substantial experience using MACCS2 and preparing Level 3 PRA models for commercial nuclear power plants in the United States. I have performed, or overseen the performance of, MACCS2 modeling in support of SAMA analyses for ten nuclear power plant sites. I also have developed similar analyses for three

proposed new reactor sites and supported reactor vendor development of MACCS2 models for new plant designs.

11. (GAT) I am providing this Joint Declaration in support of the "Applicant's Motion for Summary Disposition of the Petitioners' Contention 4 (SAMA Analysis Source Terms)." Like Dr. O'Kula, I have thoroughly reviewed the various inputs and assumptions used in FirstEnergy's original and revised SAMA analyses to calculate offsite consequences associated with a postulated severe accident at Davis-Besse, including relevant supporting documentation discussed herein. As a result, I have personal knowledge of the modeling methods, inputs, and assumptions used in the Davis-Besse SAMA analysis, as recently amended.

12. (GAT) In preparing this Joint Declaration, I also reviewed relevant pleadings of the parties and Orders issued by the Board and the Commission; relevant regulations and guidance documents; those portions of FirstEnergy's ER, as amended, describing the Davis-Besse SAMA analysis; FirstEnergy responses to NRC Staff RAIs; supporting technical documentation for the SAMA analysis prepared by FirstEnergy vendors; and various technical reports and studies.

II. ISSUES RAISED IN CONTENTION 4

13. We understand that, as admitted by the Board and narrowed in scope by the Commission, Contention 4 challenges FirstEnergy's use of the MAAP computer code to determine source terms (including release fractions) used in the Davis-Besse SAMA analysis.[5] *FirstEnergy Nuclear Operating Co.* (Davis-Besse Nuclear Power Station, Unit 1), LBP-11-13, slip op. at 50-54, 64 (Apr. 26, 2011), *aff'd in part and rev'd in part*, CLI-12-08 (Mar. 27, 2012). Specifically, Contention 4 alleges that the use of MAAP-generated source terms "appears to lead to anomalously low consequences" because: (1) the MAAP code has not been validated by the NRC; (2) the radionuclide release fractions generated by MAAP are consistently smaller for key

[5] Source term, release fraction, and other PRA-related terms used in this Joint Declaration are defined in the table in Attachment A to this Joint Declaration and discussed further throughout this document.

radionuclides than the release fractions specified in NUREG-1465, *Accident Source Terms for Light-Water Nuclear Power Plants* (Feb. 1995) (Attach. 8); and (3) other severe accident studies have observed that MAAP generates lower release fractions than those derived and used by the NRC Staff. *See* Beyond Nuclear, Citizens Environment Alliance of Southwestern Ontario, Don't Waste Michigan, and the Green Party of Ohio Request for Public Hearing and Petition for Leave to Intervene at 108-114 (Dec. 27, 2010) ("Pet.") (Errata filed Jan. 5, 2011). We address each of these three bases for the Intervenors' contention below.

III. SUMMARY OF KEY POINTS AND CONCLUSIONS

14. In this Joint Declaration, we summarize the purpose of, and methodologies required for, a PRA-based, site-specific SAMA analysis, and explain why the MAAP4[6] code used by FirstEnergy is reasonable and appropriate for that type of analysis. We also explain why Intervenors' criticisms of the MAAP code as applied in the Davis-Besse SAMA analysis are not credible. Intervenors make three principal claims in support of their contention, which, for clarity and ease of reference, we refer to as Bases 1, 2 and 3. In summary, Intervenors' contention and three supporting bases lack merit for the following principal reasons:

Basis 1: The MAAP code has been developed and maintained in accordance with NRC quality assurance standards, extensively benchmarked, applied to different reactor designs throughout the world, identified as a consensus computer code suitable for use in PRA applications, and long been accepted by the NRC for use in both safety and environmental applications, including numerous NRC-approved SAMA analyses.

Basis 2: MAAP expectedly produces source terms and release fractions that are different from, and consistently smaller than, those specified in NUREG-1465. MAAP-generated source terms serve a fundamentally different regulatory purpose and reflect modeling of different, *plant-*

[6] Although FirstEnergy used MAAP Version 4.0.6 (MAAP4) in support of the Davis-Besse SAMA analysis, we generally use the term "MAAP" below for brevity and convenience.

specific accident phenomena than those considered in NUREG-1465. NUREG-1465 was developed to define revised, *generic* accident source terms for regulatory application for future light-water reactors ("LWRs"). It postulates a release of fission products from the core of an LWR into the containment atmosphere for the purpose of calculating offsite doses in accordance with reactor siting criteria contained in 10 C.F.R. Part 100. Further, NUREG-1465 does not specify plant-specific source terms for releases from containment into the environment following a severe accident and does not take into account the source term reductions that would occur as a result of engineered and natural fission product removal mechanisms. In contrast, MAAP models the release of radionuclides from the containment *into the environment* following a postulated severe accident using plant-specific information and accounts for fission product removal mechanisms.

NUREG-1465 presents only one set of generic pressurized water reactor ("PWR") release fraction data for an *in-containment* accident source term. Applying those data to *all* 34 release categories examined in the Davis-Besse SAMA analysis, as apparently suggested by Intervenors, essentially treats all releases into the containment as releases into the environment; *i.e.*, it treats a wide spectrum of containment failure and containment bypass events equivalently. The assumption of not crediting the containment's presence, and neglecting associated passive and active engineered safety features for mitigating and delaying releases would lead to a worst-case source term scenario. It does not account for the plant-specific and probabilistic nature of a SAMA analysis, including the initiating events, accident progression and its associated likelihood, reactor core radionuclide inventory, and release fractions that are specific to each accident sequence in the plant of interest.

Basis 3: Severe accident source term estimates depend on many plant-specific design features and operational practices, and the level of technical accuracy provided by computer code

models used for source term quantification. Comparisons of MAAP to earlier studies and the source term/release fractions provided by older codes are flawed because earlier source term and release fraction estimates were based on the state of research available at the time, and the earlier severe accident codes were not as accurate as today's versions. As an example, the Intervenors cite a severe accident and risk analysis for the Zion PWR plant in a draft of the NUREG-1150 study. Pet. at 114 (citing NRC Office of Nuclear Regulatory Research, *Draft for Comment, Reactor Risk Reference Document*, NUREG-1150, Vol. 1, at 5-14 (Feb. 1987) (Attach. 9)).[7] The draft NUREG-1150 study cited by Intervenors was published over 25 years ago and involved an assessment of severe accident risks at five commercial nuclear power plants in the U.S. Davis-Besse was not one of those five plants. In addition, the state of the art for source term analysis has improved significantly since the NUREG-1150 study was performed in the mid-to-late 1980s. Intervenors' cited comparisons of MAAP-generated source terms or release fractions with those estimated over ten years earlier by different analysts for different plants—using simpler versions of other codes and different assumptions—are expected to show differences.

IV. REGULATORY AND TECHNICAL BACKGROUND

A. NRC-Required SAMA Analysis

15. Severe nuclear accidents are those in which substantial damage is done to the reactor core whether or not there are serious offsite consequences. Policy Statement on Severe Reactor Accidents Regarding Future Design and Existing Plants, 50 Fed. Reg. 32,138 (Aug. 8, 1985) (Attach. 11). In the context of a nuclear power plant PRA, a severe accident is described as a beyond design-basis accident involving multiple failures of equipment or functions. Although severe accidents generally have lower likelihoods than design-basis accidents, they may have

[7] The NRC published NUREG-1150 in final form in December 1990. NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Vol. 1, Tbl. 2.1 at 2-3 (Dec. 1990) ("NUREG-1150") (excerpts attached as Attach. 10).

greater consequences. NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," Vol. 1, at 5-1 (May 1996) ("GEIS") (Attach. 12). The NRC's GEIS provides an evaluation of severe accident impacts that applies to all U.S. nuclear power plants. *See id*. at 5-1 to 5-20. Based on the GEIS evaluation of severe accidents, 10 C.F.R. Part 51 concludes that "[t]he probability weighted consequences of atmospheric releases, fallout onto open bodies of water, releases to ground water, and societal and economic impacts from severe accidents are small for all plants." 10 C.F.R. Pt. 51, Subpt. A, App. B, Tbl. B-1 (Postulated Accidents; Severe accidents). Thus, by definition, a SAMA analysis considers postulated events whose probability of occurrence is so low that they are excluded from the spectrum of design-basis accidents postulated by NRC regulations.

16. SAMA analysis is not part of the NRC safety review for license renewal under the Atomic Energy Act ("AEA"). Rather, it is a mitigation alternatives analysis conducted pursuant to the National Environmental Policy Act ("NEPA"). The NRC's NEPA-implementing regulations in 10 C.F.R. Part 51 state that, if the NRC Staff has not previously considered SAMAs for a license renewal applicant's plant in a final environmental impact statement or in an environmental assessment, then the applicant must complete an evaluation of alternatives that may mitigate severe accidents. 10 C.F.R. § 51.53(c)(3)(ii)(L); *see also* Part 51, Subpart A, App. B, Table B-1. At the time of its 1996 Part 51 rulemaking, the Commission was unable to reach a generic conclusion regarding mitigation alternatives for purposes of license renewal, because not all licensees had completed the agency's Part 50-based ongoing regulatory program related to severe accident mitigation. Final Rule: Environmental Review for Renewal of Nuclear Power Plant Operating Licenses, 61 Fed. Reg. 28,467, 28,480-481 (June 5, 1996), amended by 61 Fed. Reg. 66,537 (Dec. 18, 1996) (Attach. 13).

17. SAMA analysis is a site-specific, frequency-weighted assessment of the benefits and costs of mitigation alternatives that might be used to reduce the risks (frequencies or consequences or both) of potential nuclear power plant severe accidents. SAMA analysis focuses on mean annual consequences and, therefore, is not intended to model a single radiological release event under specific meteorological conditions at a single moment in time. Instead, it models a set of postulated plant-specific, severe accident releases that could, based on probabilistic analysis, occur at any time under varying weather conditions during a one-year period. The objective is to estimate mean annual impacts for the entire 50-mile radius region surrounding a nuclear power plant.

18. NRC-endorsed industry guidance on SAMA analyses states: "The purpose of the analysis is to identify SAMA candidates that have the potential to reduce severe accident risk and to determine if implementation of each SAMA candidate is cost-beneficial." NEI 05-01, Rev. A, "Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document" at 1 (Nov. 2005) ("NEI 05-01") (Attach. 14); "Final License Renewal Interim Staff Guidance LR-ISG-2006-03: Staff Guidance for Preparing Severe Accident Mitigation Alternatives Analyses" (Aug. 2007) (Attach. 15) (endorsing NEI 05-01, Rev. A). A SAMA analysis thus identifies potential changes to a nuclear power plant, or its operations, that could reduce the already-low risk (frequency and/or the consequences) of a severe accident for which the benefit of implementing the change may outweigh the cost of implementation. NEI 05-01 at 1. Changes to the plant that could reduce the risk of a severe accident include plant modifications or operational changes (*e.g.*, improved procedures and augmented training of control room and plant personnel). *Id.* at 23. These potential changes are referred to as SAMAs or SAMA candidates. *Id.* at 23-26.

19. The methodology for the overall SAMA analysis approach is based on NRC guidance in NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook, Final

Report" (January 1997) (Attach. 16) and NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission, Revision 4" (August 2004) (Attach 17). Broadly speaking, a SAMA analysis involves four major sequential steps:

- Use PRAs and other risk studies to characterize the overall *plant-specific* severe accident risk by identifying and characterizing the leading contributors to core damage frequency ("CDF") and offsite risk based on a plant-specific risk study;
- Identify potential plant improvements (*i.e.*, SAMA candidates) that could reduce the risk of a severe accident;
- Quantify the risk-reduction potential and the implementation cost for each SAMA candidate; and
- Determine whether implementation of the SAMA candidates may be cost-effective.

NEI 05-01 at 2 (Attach. 14). The SAMA evaluation of a plant is based on the numerical evaluation of severe accident risk impacts in four categories: (1) offsite exposure cost, (2) offsite economic cost, (3) onsite exposure cost, and (4) onsite economic cost. *Id.* at 28.

20. Various computer codes—including the MAAP code at issue in Contention 4—are used in support of a SAMA analysis. These include codes used to develop a Level 1 PRA (analysis of initiating events and ensuing accident sequences leading to core damage) and a Level 2 PRA (analysis of accident progression leading to containment failure and bypass, and release of radionuclides to the environment). The output of the Level 1 PRA is used in the Level 2 PRA. The output of the Level 2 PRA, in turn, is used in the Level 3 PRA offsite consequences portion of the analysis. As identified in NEI 05-01, the Level 3 PRA uses the MACCS2 code to estimate the offsite dose and offsite economic impacts resulting from postulated releases of radioactive materials to the atmosphere. NEI 05-01 at 13. MACCS2 performs these calculations based on plant- and site-specific, regional, industry, and standardized regulatory inputs.

21. The basis for a SAMA analysis conducted for a U.S. nuclear power plant is a sequential, three-level PRA, *i.e.*, a comprehensive assessment of postulated accident sequences resulting in damage to the core and containment, radiological release, and their associated

frequencies. NEI 05-01 at 4 (Attach. 14). As shown in Figure 1, and as discussed above, the PRA for a commercial power reactor is divided into three levels—Level 1, Level 2, and Level 3—all of which are required to perform a SAMA analysis.

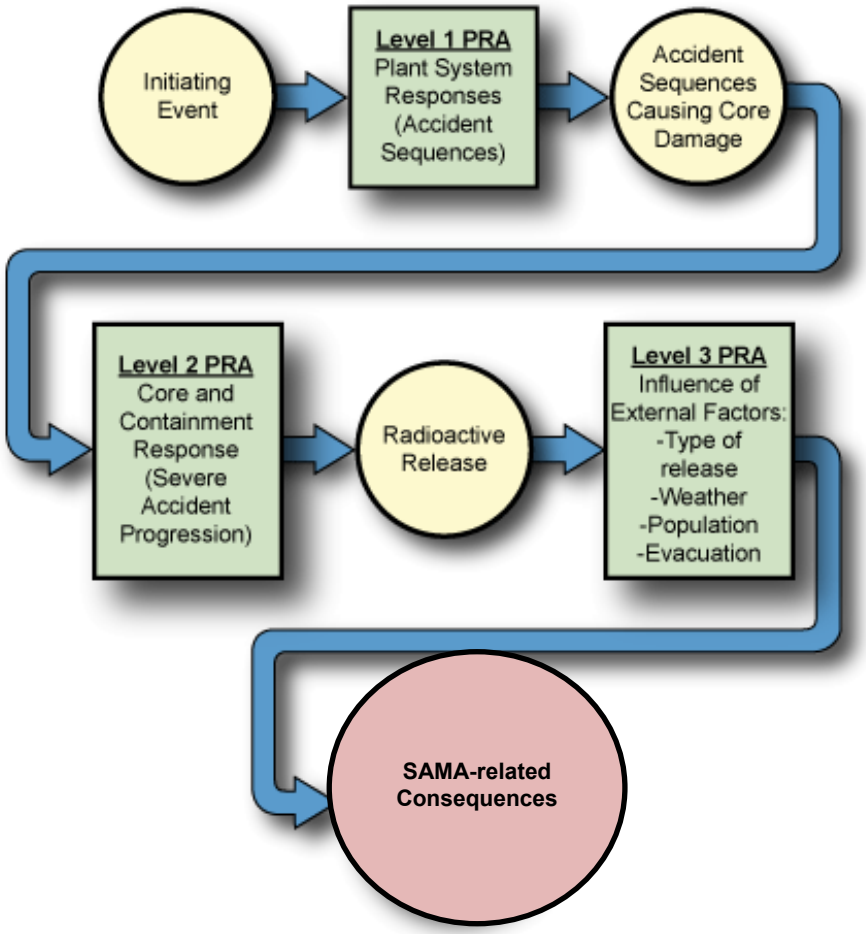

Figure 1. Three-Level Probabilistic Risk Assessment for Reactor Operation (adapted from U.S. Nuclear Regulatory Commission, *Probabilistic Risk Assessment (PRA)*. http://www.nrc.gov/about-nrc/regulatory/risk-informed/pra.html) (Attach. 18).

22. A PRA assesses the risk from operating nuclear power plants by answering three basic questions: (1) What can go wrong? (2) How likely is it? and (3) What are the consequences? The Level 3 PRA consequence analysis, performed in part with the MACCS2 code, uses the source terms and frequencies from the Level 2 PRA to estimate the 50-mile radius region offsite population dose and economic consequences, and their likelihoods. Thus, PRAs using plant-

specific information are key components of a SAMA analysis. As described further below, FirstEnergy performed a three-level PRA for Davis-Besse to support the SAMA analysis.

B. Use of the MAAP Code in NRC SAMA Analyses

23. As discussed further below, the plant-specific PRA provides, among other information, the source term information developed using the MAAP code and required as input to the MACCS2 code. MACCS2 executes three modules (ATMOS, EARLY, and CHRONC) in sequence to calculate consequence values necessary for a SAMA analysis, and models atmospheric transport and dispersion and subsequent deposition in a radial-polar grid (16 compass sectors over a 50-mile radius). NUREG/CR-6613, "Code Manual for MACCS2: Volume 1, User's Guide," at 2-1 to 2-3 (May 1998) (Attach. 19). ATMOS performs calculations pertaining to atmospheric transport, dispersion, and deposition of radioactive material, and to radioactive decay of that material both before and after its release into the atmosphere. EARLY uses the calculated air and ground concentrations, plume size, and timing information for all plume segments calculated by ATMOS and other inputs (*e.g.*, population distribution) to calculate consequences due to radiation exposure in the "emergency phase" (*e.g.*, the first seven days from the time of release). *Id*. at 2-2. CHRONC uses atmospheric transport, dispersion, and deposition information calculated by ATMOS and other inputs (*e.g.*, population distribution and economic data) to calculate (1) the long-term phase consequences due to exposure after the emergency phase (*i.e.*, the duration of the long-term exposure period is set to 30 years in most SAMA analyses); and (2) the economic impacts from each release category and the economic costs of the short-term and long-term protective actions. *Id*. at 2-2 & 2-10.

24. ATMOS requires plant-specific source term information, including the *core inventory* (*i.e.*, the amount of each radionuclide present in the reactor core at the time of accident initiation), and characteristics of the postulated release, including the amount of each radionuclide

released, release height, release duration, the sensible energy released, and other physical-chemical characteristics. The *source term* refers to the amount and isotopic composition of material released (or postulated to be released) from the reactor core during an accident. NUREG-1150, Vol. 1, 2-3 tbl. 2.1 (Attach. 10).

25. The source term may refer to radionuclide groups in the reactor core inventory at the start of an accident that are released to the containment (*i.e.*, the containment source term) or that are *released to the environment* (*i.e.*, the environmental source term). (This is an important distinction that we discuss further below in addressing the claims made in Contention 4.) An *environmental source term* describes the physical, chemical, and radiological composition of an atmospheric release. The environmental source term is used in the ATMOS atmospheric transport and dispersion module of MACCS2 to quantify the population dose and economic cost consequences that are estimated in a SAMA analysis.

26. One component of the source term is the *release fraction*, which is the fraction of the total activity of the fission products released during the postulated accident. It defines the portion of the radionuclide inventory, by radionuclide group, in the reactor core or other location, that is ultimately released. NUREG-1150, Vol. 1 at 10-4 (Attach. 10).

27. Evaluation of source terms for a SAMA analysis requires a detailed analytical model that includes a multitude of physical process sub-models that account for, among other things, the timing and performance of both passive and active plant safety features and human (*i.e.*, operator) actions affecting accident progression and containment conditions. Any radionuclide releases outside of containment are sequentially modeled from their release from the reactor core through any release path from the containment (through partial containment failure or bypass conditions), and into the environment. Source terms depend on how rapidly the accident progresses, the path by which the radionuclides escape from the reactor into containment, the path

through containment (or possibly bypassing containment altogether), and the effectiveness of both passive and active safety features, especially pools and sprays, that are intended to mitigate releases. An example of mitigation provided by sprays and pools would be the "scrubbing" or removal of radionuclides and cooling of the internal environment to which radionuclides have been released (which reduces the containment internal pressure driving the release).

28. In the U.S., source terms usually are estimated using one of two computer codes: the Methods for Estimation of Leakages and Consequences of Releases ("MELCOR") code or the MAAP code. As noted above, FirstEnergy used MAAP 4.0.6 in support of its SAMA analysis. For purposes of a Level 3 PRA/SAMA analysis, there are six parameters in the Release Description Data group of MACCS2 that need to be extracted from the MAAP code output. *See* NUREG/CR-6613, Vol. 1 at 5-23 to 5-29 (Attach. 19). Those parameters are identified in paragraph 52, *infra*.

29. MAAP simulates the dominant thermal-hydraulic and fission product phenomena in both the primary (Figure 2) and containment systems of PWRs. MAAP models have also been applied to boiling water reactors ("BWR") and other types of reactors. MAAP evaluates a broad spectrum of phenomena, including steam formation; core heat-up; cladding oxidation and hydrogen evolution; vessel failure; corium-concrete interactions; ignition of combustible gases; fluid entrainment by high-velocity gases; and fission-product release, transport, and deposition. It also addresses important engineered safety systems and allows a user to model operator interventions. Electric Power Research Institute ("EPRI") Report 1020236, "MAAP4 Applications Guidance: Desktop Reference for Using MAAP4 Software, Revision 2" at 2-2 to 2-3 (2010) ("MAAP4 Applications Guidance") (Attach. 20); Fauske & Associates, LLC, *MAAP (Modular Accident Analysis Program)*, http://www.fauske.com/pdf/MAAP.pdf (Attach. 21).

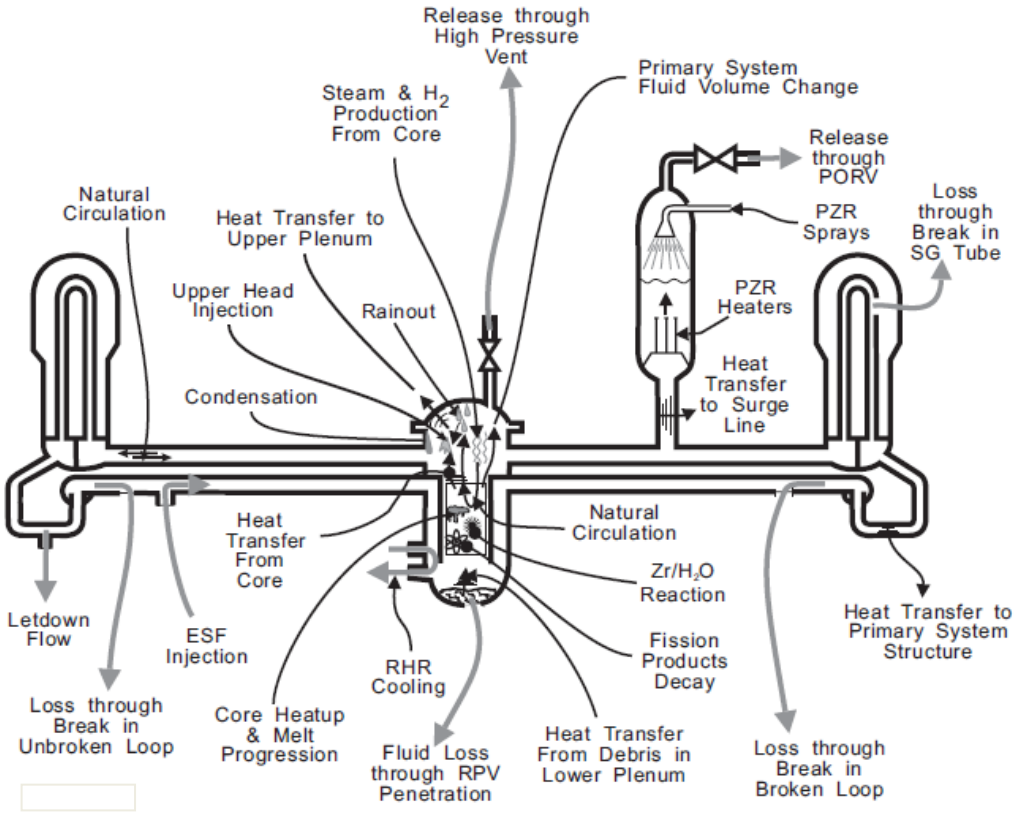

Figure 2. MAAP4 Primary System Modeling (from Fauske & Associates, LLC, *MAAP (Modular Accident Analysis Program*) (Attach. 21).

V. RESPONSE TO ISSUES RAISED IN CONTENTION 4

A. Validation of the MAAP Code (Basis 1)

30. Intervenors' claim that the MAAP code has not been "validated" by the NRC (Pet. at 108, 112, 114) is inaccurate because, as discussed later in this section, the NRC has accepted source terms calculated by numerous licensees with MAAP for use in their respective license renewal SAMA analyses. In general, a computer code in itself is not validated by the NRC, but its use for specific applications may be found acceptable for estimating certain phenomena within certain defined regimes. For example, a computer code may be used to predict the coupled thermal-hydraulic fission product transport response of reactor systems to severe accident events. If inputs and assumptions are appropriate for the computer model, and sources of uncertainty are

understood, then the results of that code may be accepted by a reviewer or regulator for purposes of the application. The NRC previously has described its approach to reviewing licensee submittals that rely on MAAP as follows:

> For each plant-specific submittal that relies on MAAP for a design-basis application, we will review those portions of the code relevant to the application, as we would any other licensing basis code. The review will generally be limited to identifying the critical MAAP models, assumptions, and code input used in the application, verifying the validity of the models by benchmarking the code with experiments and other codes, and assessing the integration of the MAAP results (*e.g.*, containment pressure and temperature history) into the analysis package. We may supplement this review by performing audit calculations (using staff codes) to confirm the results. The approval of the analysis will be limited to that specific licensing action (*i.e.*, the approval will not be an approval of MAAP.) … *This approach will also be used for plant-specific submittals that rely on MAAP for severe accident applications*, when we consider a technical review appropriate.

Letter from Gary M. Holahan, Director, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation, U.S. N.R.C., to Theodore U. Marston, Vice-President & Chief Nuclear Officer, EPRI at 1 (Dec. 4, 2001) (Attach. 22) (emphasis added). As explained below, the use of MAAP code results have been found acceptable by the NRC Staff in Level 1 and 2 PRAs, and PRA-based SAMA analyses. In these analyses, the NRC has accepted the use of the MAAP code as a tool for modeling specific severe accident phenomenology in specific reactor systems, such as a PWR's thermal-hydraulic response and fission product release characteristics under postulated accident conditions.

31. MAAP was originally developed for the Industry Degraded Core Rulemaking ("IDCOR") program in the early 1980s by Fauske & Associates, LLC (formerly Fauske & Associates, Inc.).[8] At the completion of IDCOR, ownership of MAAP was transferred to EPRI,

[8] The nuclear power industry created the IDCOR program in response to the 1979 accident at Three Mile Island Unit 2 ("TMI-2") to independently evaluate technical issues related to potential severe accidents at LWR nuclear power plants. IDCOR's original mission was to gather and critically review existing technical work related to the severe accident issues and to perform the additional technical work required to develop a comprehensive

which was charged with maintaining and improving the code. Starting in the late 1980s, the MAAP3B version became widely used, first in the United States and then worldwide, to support success criteria determination, human action timing evaluations, and Level 2 analyses for Individual Plant Examinations ("IPEs") required by the NRC. MAAP3B was updated to MAAP4 in the mid-1990s to expand its modeling capabilities. MAAP4 incorporates updated physical models for core melt, reactor vessel lower head response, and containment response that provide improved mechanistic modeling of severe accident phenomena. Several organizations, including EPRI and the DOE, sponsored the development of MAAP4. As part of the development process, a committee of independent experts reviewed MAAP4 to ensure that it is state-of-the-art and applicable for accident management evaluations. Further, a Design Review Committee comprising senior members of the nuclear safety community reviewed the new software. MAAP4 Applications Guidance at 2-2 (Attach. 20).

32. EPRI licenses MAAP to a wide array of entities, such as utilities, vendors, and research organizations, including universities. The majority of MAAP users are members of the MAAP Users Group ("MUG"). The MUG provides direction and funding for code maintenance, enhancements, and benchmarking; facilitates information transfer through biannual meetings and the issuance of various communications on code problems and best practices; and supports industry and regulatory acceptance. MAAP4 Applications Guidance at 2-2 (Attach. 20).

33. Fauske & Associates is the current maintenance contractor for the code. MAAP and its successor versions, including MAAP4, were developed in accordance with 10 C.F.R. Part 50, Appendix B and International Organization for Standardization ("ISO") 9001 quality assurance requirements. MAAP4 Applications Guidance at 2-2 (Attach. 20). EPRI has identified the MAAP code (versions 4.0.5 and later) as a "consensus model" suitable for use in evaluation of

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understanding of these issues. The IDCOR program also served as the industry interface with the NRC on these matters.

PRA success criteria. EPRI Report 1013492, "Probabilistic Risk Assessment Compendium of Candidate Consensus Models" at 2-3 (2006) (Attach. 23).

34. MAAP and MELCOR have been sponsored by the industry and the NRC, respectively, in part to predict the phenomenology of severe accident progression and have been benchmarked, to the extent possible, with applicable severe accident experimental research results. For example, MAAP has been successfully benchmarked against numerous severe accident studies and the Three Mile Island Unit-2 ("TMI-2") core melt accident. The extensive benchmarking of MAAP is documented in the Section 7 (MAAP Benchmarks) and Appendix F (Summaries of MAAP Benchmarks) of EPRI's MAAP4 Applications Guidance, and also in a 2007 report issued by the Nuclear Energy Agency ("NEA"). The 2007 NEA report summarizes key MAAP benchmarking activities as follows:

> Many comparisons between the MAAP code and separate effects tests, integral experiments, actual plant transients, and accidents have been performed to illustrate the performance of individual models and to provide confidence in the MAAP integral results. The assessment matrix listed … shows the experimental benchmarking status of the MAAP computer code. It is seen that the various code versions … have been compared to several separate effects and integral experiments. These include: CORA and PHEBUS (core damage); LOFT FP-2 (integral severe accident test); ABCOVE (aerosol behaviour); CSE (containment spray); COPO (molten pool heat transfer); FARO (debris quenching); Surtsey IET (DCH); SWISS, SURC-4, ACE, KfK BETA (core-concrete interaction); NUPEC mixing tests; Marviken, FAI, and GE vessel blowdown tests; and HDR containment experiment, among many others. The current version of the code, MAAP4, has also been benchmarked against the TMI-2 accident. This comparison study shows that MAAP4 provides a reasonable simulation of the TMI-2 accident in terms of the system response prior to core uncovery, during core degradation, following core reflood, and the lower head behaviour after 224 minutes. These are all severe accident processes that are essential for application of computer codes for decisions related to design, operations, emergency operating procedures, and accident management.

NEA Committee on the Safety of Nuclear Installations, NEA/CSNI/R(2007)16, *Recent Developments in Level 2 PSA and Severe Accident Management* at 36 (Nov. 2007) (Attach. 24).

35. Furthermore, the MAAP code has long been used by nuclear power plant licensees and other entities to predict the responses of nuclear power plants during postulated severe accidents. To our knowledge, MAAP is the most commonly used code in the U.S. for such purposes.[9] The use of MAAP and its successor versions in IPEs and subsequent PRA applications has been accepted by the NRC Staff for many years, including the code's use in design and licensing basis applications.[10] Moreover, the MAAP code has been used throughout the world and produces results comparable to MELCOR, a similar code developed by Sandia for the NRC for use in modeling severe accidents and performing PRAs.

36. Although SAMA analysis is a NEPA-related requirement (as opposed to a safety-based analysis subject to 10 C.F.R. Part 50 requirements), numerous license renewal applicants have used the MAAP code to support NRC-approved SAMA analyses, including several very recent recipients of renewed operating licenses. *See*, *e.g.*, NUREG-1437, Supp. 47, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Columbia Generating Station – Final Report," Vol. 2, App. F at F-2, F-6 to F-7, F-27 (Apr. 2012) (Attach. 27); NUREG-1437, Supp. 45, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Regarding Hope Creek Generating Station and Salem Nuclear Generating Station, Units 1 and 2, Vol. 2, App. G at G-4, G-6, G-15 to G-16 (Mar. 2011) (Attach. 28).

37. In view of the above, it is clear that the MAAP code has been subject to extensive benchmarking and technical review by the nuclear safety community. In addition, the NRC has

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[9] *See also* Kenneth D. Kok, Ed., *Nuclear Engineering Handbook* at 539 (2009) (Attach. 25) ("The most commonly used Level-II PRA tools include CAFTA for fault tree analysis … and the modular accident analysis program (MAAP) for severe accident simulation.").

[10] For example, the Staff accepted the use of MAAP in its 1994 design certification approval for the Advanced Boiling Water Reactor ("ABWR") design in NUREG-1503, "Final Safety Evaluation Report Related to Certification of the ABWR Reactor Design," Vol. 1 at 19-53 to 19-55 (July 1994) (Attach. 47), and has done so for other subsequent design certification approvals. *See*, *e.g.*, NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," Vol. 1 at 19-61 (Sept. 2004) (Attach. 48) (finding the applicant's use of both the MAAP4 and MACCS2 codes "to be consistent with the present state of knowledge regarding severe accident modeling" and acceptable).

reviewed applicants' use of MAAP in various licensing contexts and found it to be acceptable for severe accident modeling and probabilistic risk assessment. The code's use as a basis for fission product release from the core, transport into the containment, and subsequent environmental source term prediction is reasonable and appropriate under NEPA and 10 C.F.R. Part 51. In our view, Intervenors have offered no credible information to support a different conclusion. In fact, Intervenors' position runs counter to the international nuclear community's recognition of MAAP as a state-of-the art code and to the NRC's acceptance of the code for use by its licensees in both safety and environmental applications, including many SAMA analyses.

B. Differences in MAAP-Generated and NUREG-1465 Source Terms (Basis 2)

1. The NUREG-1465 Source Term Represents Only Radionuclides Released into the Containment Atmosphere as a Result of a Core-Melt Accident

38. Intervenors' claim that NUREG-1465 source terms are more appropriate for use in a SAMA analysis than plant-specific, MAAP-generated source terms is not valid. Pet. at 112. As an initial matter, one must recognize that the reactor accident source term generally serves two purposes in the U.S. nuclear regulatory process. F. Eltawila, NRC, "NRC Source Term Research – Outstanding Issues and Future Directions," European Review Meeting on Severe Accident Research, Karlsruhe, Germany, June 12-14, 2007, Slide 2 ("Eltawila") (Attach. 28). The first purpose is for licensing, safety analysis, and regulatory compliance, particularly in meeting 10 C.F.R. Part 100 reactor siting requirements. *Id.* For this purpose, a source term representing the release of radioactive materials into the reactor containment is used to assess the adequacy of reactor containments and engineered safety systems, as well as the environmental qualification of equipment inside the containment that must function following a design-basis accident. This source term also is used to show that dose criteria at the exclusion area boundary are met by assuming the maximum allowable design leak rate from the containment. *See* 10 C.F.R. § 50.34(a)(1)(ii)(D) and § 100.11. *NUREG-1465 source terms are applicable for this purpose.*

39. The second purpose for which a reactor accident source term is developed is to simulate a release of radioactive material to the environment (*i.e.*, outside containment) following a hypothetical reactor accident. *See* Eltawila, Slide 2 (Attach. 28). This second source term is input to models of radionuclide environmental transport and dispersion and accident consequences that, among other purposes, are used for Level 3 PRAs and SAMA cost-benefit analyses, which are best-estimate analyses. The use of the MAAP-based source term associated with releases to the environment for the Davis-Besse PRA and its SAMA analysis supports this latter purpose; *i.e.*, it is a crucial element of Level 3 PRA and SAMA cost-benefit analyses.

40. The inapplicability of NUREG-1465 to SAMA analysis is evident from the origin and stated purpose of that report. In 1962, the Atomic Energy Commission published TID-14844, "Calculation of Distance Factors for Power and Test Reactors" (Mar. 1962) (Attach. 29), which specified a release of fission products from the core to the reactor containment in the event of a postulated accident involving a substantial meltdown of the core. This "source term," the basis for NRC Regulatory Guides 1.3 and 1.4, has been used to determine compliance with the NRC's Part 100 reactor siting criteria, and to evaluate other important plant performance requirements.

41. In the period between TID-14844 and the issuance of NUREG-1465, the knowledge base for severe LWR accidents and the associated behavior of released fission products was substantially updated and augmented. The NRC developed and issued NUREG-1465 in 1995 "to provide a postulated fission product source term *released into containment* that is based on current understanding of LWR accidents and fission product behavior." NUREG-1465 at vii (Attach. 8) (emphasis added). NUREG-1465 states that its "primary objective … is to define a revised accident source term for regulatory application for *future* LWRs." *Id.* at 3 (emphasis added).

42. A SAMA analysis uses the plant-specific Level 2 PRA analyses to postulate and model the release of radionuclides into the environment during a severe accident. In contrast, the NUREG-1465 generic source term solely represents radionuclides released into the containment (as contrasted to the environment); *i.e.*, it assumes a "release resulting from 'substantial meltdown' of the core into the containment . . . [and assumes] that the containment remains intact but leaks at its maximum allowable leak rate."[11] NUREG-1465 at 1 (Attach. 8) . Indeed, NUREG-1465 states: "In this document, a release of fission products from the core of a light-water reactor (LWR) *into the containment atmosphere* ("source term") was postulated for the purpose of calculating off-site doses in accordance with 10 CFR Part 100, 'Reactor Site Criteria.'" *Id.* at vii (emphasis added). Notably, a November 2002 report cited by the Intervenors in their Petition (Pet. at 114) further confirms that NUREG-1465 postulates a release of fission products from the core of an LWR into the containment atmosphere, not to the environment.[12]

2. The NUREG-1465 Source Term Does Not Describe the Release of Radionuclides to the Environment as Postulated in a SAMA Analysis

43. It should be expected that MAAP produces release fractions that are different from, and generally smaller than, the release fractions specified in NUREG-1465. As discussed above, MAAP models the release of radionuclides from the containment into the environment following a postulated severe accident. In contrast, the NUREG-1465 source term describes the amounts and types of radioactive material that would enter the containment. The NUREG-1465 source term

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[11] Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Jan. 2000) (Attach. 30), provides environmental source terms to be applied for *design basis accident* consideration of the NUREG-1465 source term, and prescribes the performance of engineered safety features and the containment leak rate. It also acknowledges that "NUREG-1465 presents a representative accident source term for a boiling-water reactor (BWR) and for a pressurized-water reactor (PWR). These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the *release to the containment*." (emphasis added).

[12] *See* Energy Research, Inc., ERI/NRC 02-202, "Accident Source Terms for Light-Water Nuclear Power Plants: High Burnup and Mixed Oxide Fuels" at 5 (Nov. 2002) (Attach. 46). The report states that the "representative" PWR and BWR source terms in NUREG-1465 "are characterized by the composition and magnitude of fission product release *into containment*, the timing of the release *into containment*, and the physical and chemical forms *in containment*." *Id.* (emphasis added).

does not specify the source term released from containment into the environment following a severe accident. Nor does it take into account the reductions of the source term that would occur due to fission product removal mechanisms.

44. In short, the amount of radioactive material that enters the containment atmosphere is different from (*i.e.*, larger than) the amount of radioactive material that enters the environment as a result of a severe accident. Although NUREG-1465 recognizes the importance of fission product removal mechanisms, including engineered safety features ("ESFs") and natural processes (*e.g.*, aerosol deposition and the sorption of vapors on equipment and structural surfaces), it does *not* consider the effects of such mechanisms (*e.g.*, containment sprays, aerosol deposition) in the numerical estimates of source terms. NUREG-1465 at 17-21 (Attach. 8). That is, NUREG-1465 does not provide numerical estimates of the containment source terms that account for the effects of in-containment fission product removal mechanisms (*e.g.*, containment sprays, aerosol deposition). Rather, it directs the reader to use appropriate methodologies in crediting fission product removal or reduction within containment. NUREG-1465 at 4-5, 17-18. In contrast, MAAP does model and credit these ESFs as fission product removal mechanisms. *See* ¶ 31 & Fig. 2, *supra*. The developer of MAAP has noted this fact:

> Due to the strong dependence of fission product retention of plant specific features and accident sequence progression, however, NUREG-1465 source terms do not already credit retention. This is left up to the individual licensees.
> …
> The advantage of using [MAAP] is that, in a single integrated analysis, it will provide time dependent fission product release from the core, transport to the containment, leakage to the reactor or auxiliary buildings, credit for all major engineered safeguard features, and modeling of all active and passive fission product retention mechanisms.

Fauske & Associates, Inc., Technical Bulletin No. 1295-1, "BWR MSIV Leakage Assessment: NUREG-1465 vs. MAAP 4.0.2," at 1 (Attach. 31). Thus, we would expect to see obvious

differences between a source term into the containment and a source term from the containment into the environment.

3. As a Best-Estimate Engineering Evaluation that Seeks to Quantify Risk, the Davis-Besse SAMA Analysis Uses PRA Methods and Requires Plant-Specific Source Term Information

45. NUREG-1465 provides generic, default source terms. However, PRA and SAMA analyses are best-estimate engineering evaluations that seek to maximize the use of plant-specific data. Use of a thermal-hydraulic code, like MAAP, to develop plant-specific release fractions for a SAMA analysis is strongly preferred and technically superior to using generic inputs from other sources, such as the NUREG-1465 or NUREG-1150 reports cited by Intervenors. Use of the plant-specific inputs in the SAMA analysis allows for better resolution of data and more accurate portrayal of plant-specific response to postulated severe accident phenomenology, and better serves the purpose of evaluating the benefits of potential plant improvements.

46. Intervenors' suggestion that FirstEnergy use NUREG-1465 generic source term values, which do not account plant-specific differences, is directly contrary to established NRC studies and guidance documents that have informed countless PRAs and SAMA analyses. *See*, *e.g.*, Final Rule, Environmental Review for Renewal of Nuclear Power Plant Operating Licenses, 61 Fed. Reg. 28,467, 28,480 (June 5, 1996) (Attach. 13) (referring to SAMA analysis as "site-specific" and stating that the Commission expects that significant efficiency can be gained by using site-specific individual plant examination (IPE) and individual plant examinations of external events (IPEEE) results in the consideration of severe accident mitigation alternatives); NUREG-1150, Vol. 1, at 1-3 ("One of the clear perspectives from this study of severe accident risks and other such studies is that characteristics of design and operation specific to individual plants can have a substantial impact on the estimated risks."); Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific

Changes to the Licensing Basis," Rev. 2 at 7 (May 2011) (Attach. 32) (stating that the scope, level of detail, and technical acceptability of these risk-informed analyses are to "be based on the as-built and as-operated and maintained plant" and "reflect operating experience at the plant."); NEI 05-01 at 2 (Attach. 14) (suggesting use of "plant-specific" PRA models in SAMA analyses).

47. Plant-specific source terms developed for SAMA analysis must consider a spectrum of probabilistically-significant accident scenarios to have any meaning from a risk quantification perspective. As discussed earlier, the progression from the failure of individual plant components to the determination of accident frequencies, accident progressions, and offsite consequences involves plant- and site-specific phenomena and can be separated into the three PRA levels. The Level 1 PRA establishes the plant damage states and frequency of reactor core damage frequency or CDF. The Level 2 PRA determines different accident progressions and a set of radioactive release conditions from the containment that are assigned to similar representative groups (release categories). The Level 2 PRA defines the sequence of events resulting in a radioactive release to the environment. The source term analysis then follows and quantifies the amount of radioactivity released for a given sequence and the frequency of occurrence (*i.e.*, release categories and their respective frequencies). The Level PRA 3 combines the Level 2 PRA results with site-specific parameters (*e.g.*, population distribution, meteorological data, land use data, and economic data) for the Level 3 PRA to estimate offsite public dose and offsite economic consequences of those releases to the environment.

48. This is precisely the approach that FirstEnergy used in performing the Davis-Besse SAMA analysis. As discussed in ER Sections E.3.1, E.3.2, and E.3.3, FirstEnergy used the Davis-Besse Level 1 PRA and Level 2 PRA models to estimate the CDF and release category frequencies, as well as the source terms, for use in the SAMA analysis. Fault tree and containment event tree ("CET") logic models, plant data, and mechanistic models of severe

accident phenomena (*e.g.*, MAAP) were used as part of this process. The Level 1 PRA included initiating event ("IE") and core damage ("CD") sequence analyses and yielded a set of plant damage states ("PDS") and associated frequencies.[13] The Level 2 PRA used CET and deterministic source term models to provide a set of 34 release categories, each of which has a characteristic frequency and unique timing and fission product magnitude characteristics that represent the release to the environment. The 34 release categories determined from the Level 2 Davis-Besse PRA, based in part on the MAAP analysis, were applied in the MACCS2 SAMA analysis, along with other site-specific inputs, to calculate the Davis-Besse offsite population dose risk (in units of person-rem/year) and offsite economic cost risk (in units of dollars/year), the key risk metrics in a SAMA analysis. Figure 3 illustrates the sequential analyses that are performed as part of a three-level PRA and include the use of the MAAP and MACCS2 codes.

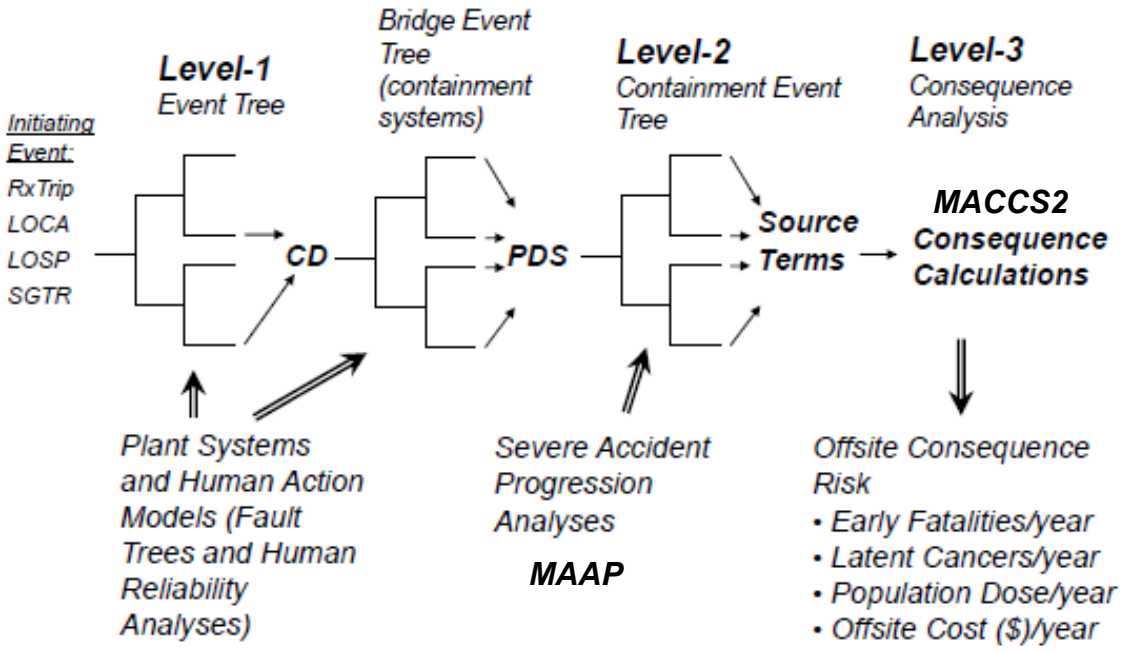

Figure 3. Sequential Analyses Performed as Part of a Three-Level PRA (based on D. Harrison, NRC, Chief, PRA Licensing Branch, "Perspectives on PSA Technology and Applications," Slide 5, *Fire PRA China PSA Workshop*, January 10, 2010 (Attach. 33).

[13] Initiating events may include, for example, a plant trip, loss-of-coolant accident, loss of offsite power, or steam generator tube rupture.

4. Plant-Specific, MAAP-Generated Source Terms Are Integral to the Davis-Besse SAMA Analysis

49. The specifics of the source terms required for a SAMA analysis are highly dependent upon the specifics of the accident progressions themselves. For that reason, PRA analysis uses detailed design-, plant type-, and site-specific information to identify initiating events and their likelihood of potentially leading to core damage, and to establish the CDF, subsequent reactor containment release, and environmental release conditions. The methodology used to develop source terms for a SAMA analysis must account for plant-unique conditions, plant design, support system dependencies, plant maintenance and operating procedures, operator training, and the interdependencies among these factors that can influence the CDF estimate for a specific plant.

50. MAAP meets this important requirement. It is an integral code that treats the full spectrum of important phenomena that could occur during an accident, simultaneously modeling those that relate to the thermal-hydraulics and to the fission product transport and deposition. It also simultaneously models the primary system and the containment (including the influence of mitigative systems and the effects of operator actions).

51. The MAAP 4.0.6 calculations provide a deterministic analysis of Davis-Besse under postulated severe accident conditions for a variety of initiating events and include the influence of operator actions and safety system actuation on accident sequence progression. The MAAP 4.0.6 calculations predict the integrated response of the reactor core, primary system, steam generators, and primary containment building. Results include the time of core damage and reactor vessel failure to support Level 1 PRA success criteria, as well as containment response and fission product source term characterization to support the Level 2 and Level 3 assessments. For each of the 34 release categories examined in the Davis-Besse SAMA analysis, a representative MAAP case was used to estimate (1) the timing of the radioactive release into the environment

and (2) the magnitude of the radioactive release into the environment. The source term defined for each release category from the Level 2 PRA was processed in the MACCS2 code.

52. Among other inputs to MACCS2, the input parameters require output information extracted from MAAP, and the development of the core inventory. Specifically, there are six input variables required by MACCS2 that come from the output of MAAP: (1) the time after accident initiation that the offsite alarm is initiated (OALARM), (2) the heat content of release segment (PLHEAT), (3) the height of the plume segment at release (PLHITE), (4) the duration of release (PLUDUR), (5) the time of release for each plume (PDELAY), and (6) the release fraction for each radionuclide group (RELFRC). The core inventory for the Davis-Besse Level 3 PRA was obtained from plant-specific calculations performed using the ORIGEN-2 code. For conservatism, the Davis-Besse core inventory was evaluated at the 24-month end-of-cycle for all 177 fuel assemblies. This assumption is generally conservative because at the end-of-cycle, the radionuclide quantities in the core would be at their peak levels for the 24-month cycle. In total, the activity levels of 58 radionuclides (represented in nine fission product groups) were evaluated as part of the Davis-Besse reactor core inventory for subsequent analysis in the MACCS2 code.

5. Use of Generic Source Terms from NUREG-1465 is Not Justified and Would Inappropriately Distort the SAMA Analysis Results

53. NUREG-1465 presents only one set of PWR release fraction data and one set of BWR release fraction data. NUREG-1465 at 13 tbl. 3.13 ("PWR Releases Into Containment") and tbl. 3.12 ("BWR Releases Into Containment") (Attach. 8). Use of the NUREG-1465 source term as a surrogate for the release into the environment instead of the Davis-Besse, plant-specific Level 2 PRA, which develops accident-specific release categories for input to the consequence analysis for the SAMA analysis, leads to an overly conservative estimate and lacks technical merit. It essentially treats all types of postulated core melt accident releases into the containment as releases into the environment; *i.e.*, it treats containment failure sequences and containment by-pass

events equivalently. The assumption of not crediting the containment's presence, and neglecting associated passive and active engineered safety features for mitigating and delaying releases, leads to one worst-case source term scenario. This magnitude of release is only PWR or BWR-specific, and does not quantify the effects of plant-specific features for which a SAMA analysis provides a reasonable, NEPA-compliant, cost-benefit analysis evaluation. Thus, using the NUREG-1465 source term instead of plant-specific information from the Level 1 and Level 2 PRA for a given plant would oversimplify the SAMA cost-benefit process and likely lead to technically unfounded conclusions about a particular plant's offsite risks.

54. As proposed by Intervenors, NUREG-1465 PWR source term data in the Davis-Besse SAMA analysis would be applied to *all* 34 release categories; *i.e.*, from containment bypass-steam generator tube rupture (RC 1) source terms through no-failure, containment maintained intact with design leakage (RC 9) source terms. However, for Davis-Besse, approximately 90% of the core damage sequences involve accidents in which the containment retains its structural integrity (*i.e.*, radiological release is limited to containment leakage, as modeled in RC 9.1 and 9.2), and the remaining 10% would be the result of early containment failure and other events (*e.g.*, containment bypass events, specifically steam generator tube rupture and interfacing system loss of coolant accidents). Additionally, early containment failure and containment bypass are different event types, with significant differences in sequence progression, timing, release pathways, and fission product deposition and removal mechanisms. These different event types logically would result in different source terms and release fractions. Use of NUREG-1465 release fractions for all release categories would essentially treat all releases into the containment as releases into the environment and greatly distort the results of the SAMA analysis. Indeed, not crediting the containment's presence and neglecting associated passive and

active engineered safety features for mitigating and delaying releases would lead to a worst-case source term scenario without any technically supported weighting by likelihood of occurrence.

55. It also bears mention that an integrated computer code such as MAAP may serve multiple functions in a Level 2 PRA. FirstEnergy used the MAAP code to support the entire Davis-Besse PRA that serves as a major input to the SAMA analysis. For example, FirstEnergy used MAAP to support the development of plant equipment success criteria (*e.g.*, amount of flow required to meet core cooling needs at specific times) and to develop timelines for operator actions to determine human error probabilities included in the PRA. Use of alternate data for release fractions as inputs to the Level 3 analysis does not obviate the dependence of the Davis-Besse SAMA analysis on the MAAP code.

56. In summary, the distinct phenomenological bases and regulatory purposes of the NUREG-1465 and MAAP source terms explain the relative numerical differences in the amount of radionuclides and the timing for the release. Due to containment ESFs (*e.g.*, containment air coolers, containment spray) and natural depletion processes (*e.g.*, aerosol deposition and containment holdup), the source term released from the reactor coolant system into containment expectedly is different from that of the containment into the environment. Thus, the NUREG-1465 and MAAP source terms *should* differ, with the MAAP source term being the smaller of the two.

57. Use of an overstated source term from NUREG-1465 would have numerous (and unjustified) effects. Such effects include exaggerated early and long-term health effects, incorrect determination of the size of the area that might become contaminated, inflated offsite economic losses, and incorrect estimates of the dollar value of SAMA candidates. The net effect would be to distort the SAMA process, and likely misrepresent the risk reduction effectiveness of plant-

specific SAMA candidates. Such SAMA candidates would be technically unmerited because they arise from applying a generic source term basis.

C. Inapplicability of Historical Release Fraction Comparisons Cited by Intervenors (Basis 3)

58. In Contention 4, the Intervenors also cite historical comparisons between release fractions developed using earlier versions of the MAAP code and release fractions developed using other codes. The first is a 1987 draft of the NUREG-1150 severe accident risk study that, in examining accident risk at Zion Nuclear Station, found that "the MAAP estimates for environmental release fractions were significantly smaller" than those obtained with the Source Term Code Package[14] ("STCP") computer code (the primary code used in the NUREG-1150 study). Draft NUREG-1150, Vol. 1 at 5-14 (Attach. 9). The second is a 2002 Brookhaven National Laboratory ("BNL") report reviewing combustible gas control availability at ice condenser and Mark III containment plants. The BNL report compared the Level 2 portion of the PRA results for the Catawba plant (obtained using the MAAP code) with a "typical NUREG-1150 release" for the Sequoyah plant (obtained using the STCP and MELCOR codes). John R. Lehner et al., Brookhaven National Laboratory, "Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice Condenser and Mark III Containment Plants, Final Letter Report" at 17 (Dec. 2002) ("BNL report") (Attach. 34). The BNL report states that the "NUREG-1150 release fractions for the important radionuclides are about a factor of 4 higher than the ones" in the Catawba PRA, and that the "differences in the release fractions . . . are primarily attributable to the use of the different codes in the two analyses." *Id.*

[14] As noted on the Sandia MELCOR website (http://melcor.sandia.gov/), the STCP is the predecessor to MELCOR: "MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light-water reactor nuclear power plants. MELCOR is being developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission as a second-generation plant risk assessment tool and the successor to the Source Term Code package."

59. Neither of the Intervenors' comparisons is germane to FirstEnergy's use of MAAP-generated source terms or release fractions in the Davis-Besse SAMA analysis. Although its remains a seminal document, the final NUREG-1150 study was completed over 20 years ago and involved an assessment of the risks from severe accidents at five commercial nuclear power plants in the United States. Davis-Besse was not one of those five plants. Furthermore, the IDCOR (MAAP) to NUREG-1150 (STCP) comparison of Zion results cited by Intervenors was only one of four sets of plant results compared in the February 1987 draft of NUREG-1150 (with several other comparisons in the draft report showing reasonable agreement). In addition, after extensive peer review of, and public comment on, the February 1987 draft, NUREG-1150, Volume 1, was issued as a second draft in 1989, before being published as a final report in December 1990. In summary, the report and its underlying technical analyses were substantially modified in two rounds of review before the report's final publication in December 1990. Notably, one of the changes included deleting the specific discussion comparing the MAAP and STCP results for Zion, such that the comparison cited by Intervenors in Contention 4 was not incorporated into the final December 1990 version of NUREG-1150.

As discussed previously, severe accident source term estimates from computer codes depend on user assumptions and expertise, the extent to which plant-specific passive and active design features are modeled, the degree of benchmarking,[15] and the technical accuracy provided by computer code models and their underlying algorithms. While the final 1990 NUREG-1150 report still is relevant to the nuclear safety community's understanding of severe accident progression, additional severe accident research performed in the U.S. and abroad in the 25 years since the 1987 draft of NUREG-1150 was issued has significantly improved that understanding.

[15] In this context, benchmarking refers to comparison of code predictions with experiments, other qualified codes that model the same phenomena and, in some situations, hand or spreadsheet calculations.

One of these research efforts, a recent NRC-sponsored project, is discussed later in this Joint Declaration (Paragraphs 65 to 69).

60. In using STCP in the NUREG-1150 study to predict complex phenomena, the study's authors noted that they used both expert elicitation and additional computer codes to augment the results of simplified STCP models. For example, with respect to the core degradation process, the NUREG-1150 authors stated the thermal-hydraulic model in the STCP "uses simplified models and assumptions for the treatment of some of the very complex steps in the core degradation process, such as fuel slumping into the lower plenum of a reactor vessel." NUREG-1150, Vol. 3, App. D at D-17 (Attach. 10). More realistic models such as MELCOR and MAAP were used to adjust the thermal-hydraulic estimates affecting core degradation, ultimately leading to differences in the source term. *Id.*

61. In accounting for the effect of more realistic models evaluating the same sequences and for the same basis plant (Zion), Henry and Rahn (2004) showed reduced environmental source terms comparing PWR sequences from WASH-1400 (1975)[16] to updated analyses in NUREG-1150 (1990) and in an updated analysis performed using MAAP. Frank J. Rahn and Robert E. Henry, "Release and Dispersion of Radioactivity from Reactor Fuel Research and Analytical Results Leading to Reductions in Radiological Source Terms," American Nuclear Society Position Statement No. 65 on "Realism in the Assessment of Nuclear Technologies," Appendix 1A (June 2004) (Attach. 36). To illustrate the time available for corrective action and the mitigation of fission product releases, Henry and Rahn evaluated large dry type accident sequences (Station Blackout or "SBO") SBO sequences (*i.e.*, AC power recovery at 12 hours and one without power recovery). The Henry and Rahn results showed that there are substantial processes (*e.g.*, heat transfer and fission product chemistry) within the reactor coolant system and the containment that

[16] WASH-1400 (NUREG-75/014), "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants" (1975) (excerpt attached as Attach. 35).

extend the interval before releases to the environment occur and also substantially limit the magnitude of releases to the environment for severe core damage accidents. In summary, the comparison showed lower release fractions from the containment to the environment for NUREG-1150 and MAAP large dry source terms than comparable (PWR 3 and 7) WASH-1400 source terms and that the MAAP-based release fractions to the environment were about the same or slightly higher when compared to similar, corresponding sequences from NUREG-1150.

62. Regarding the 2002 BNL report's comparison of the Catawba and Sequoyah plants' release fractions, it should be noted that both of these plants have ice condenser containments, while Davis-Besse has a dry, ambient pressure containment type. In any case, in its Supplemental Environmental Impact Statement for Catawba license renewal, the NRC Staff compared similar sequences between the two studies—NUREG-1150 and Revision 2b of the Catawba PRA,[17] which included the plant's IPE models—and concluded there was "reasonable agreement" for the closest corresponding release scenarios. NUREG-1437, Supp. 9, at 5-9 to 5-10 (Attach. 37). Specifically, the Staff provided the following summary:

> The Staff reviewed the process used by Duke to extend the containment performance (Level 2) portion of the IPE to the offsite consequence (Level 3) assessment. This included consideration of the source terms used to characterize fission product releases for each containment release category and the major input assumptions used in the offsite consequence analyses. This information is provided in Section 6.3 of Duke's IPE submittal. Duke used the Modular Accident Analysis Program (MAAP) code to analyze postulated accidents and develop radiological source terms for each of 29 containment release categories used to represent the containment end-states. These source terms were incorporated as input to the MACCS2 analysis. The staff reviewed Duke's source term estimates for the major release categories and found these predictions to be in reasonable agreement with estimates of NUREG-1150 (NRC 1990) for the closest corresponding release scenarios. The staff concludes that the assignment of source terms is acceptable. *Id.* at 5-10.

[17] Letter from Gary R. Peterson, Vice President, Duke Energy Corporation, to Document Control Desk, U.S. N.R.C., Attach. 1, "Catawba PRA Revision 2b Summary Results" (Apr. 18, 2001).

63. The state of the art for source term analysis has improved considerably since the NUREG-1150 study was performed in the 1980s. This is a well-known fact that Intervenors' faulty comparisons fail to consider. For example, the comparison between the Catawba Level 2 PRA release fractions and the NUREG-1150 Sequoyah release fractions represents a difference of more than ten years in terms of severe accident modeling (~2002 versus ~1990). Also, the comparison uses a release category that represents an early containment failure in which the Catawba source term is based on an "early containment failure without ex-vessel release" assumption that may not have been applied in the Sequoyah source term. *See* Memorandum from Asimios Malliakos, Probabilistic Risk Analysis Branch, Division of Risk Analysis and Applications, Office of Nuclear Regulatory Research, to Marc A. Cunningham, Chief, Probabilistic Risk Analysis Branch, Division of Risk Analysis and Applications, Office of Nuclear Regulatory Research, "Telecommunication with Duke Energy Corporation in Support of Generic Safety Issue (GSI) 189, 'Susceptibility of Ice Condenser and BWR Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident,'" Attach. 1, 3 (Oct. 8, 2002) (Attach. 38). Early containment failure, in this case, seems to be associated with high pressure in the Reactor Coolant System at reactor vessel failure, with the resulting blowdown dispersing the corium into the lower containment area. With the debris bed spread over a large area, the debris bed will be coolable, preventing the ex-vessel release of fission products, such as that due to molten core-concrete interactions, and subsequently leading to a smaller source term to the environment. This assumption apparently was not applied in the earlier NUREG-1150 analysis for Sequoyah. Although Davis-Besse also is a PWR, it has a dry, ambient air containment type, whereas both Catawba and Sequoyah are typical of ice condenser containment PWR plants.

64. Since the issuance of NUREG-1150, better understanding of heat transfer and removal from the reactor pressure vessel ("RPV") during severe accident sequences; improved

insights into iodine, cesium, and other fission product group chemistry from contemporary research; and modeling improvements suggest that the early containment failure releases potentially could be smaller than previously concluded. *See* ¶ 61, *supra*. Thus, the BNL report's comparison of MAAP-based source terms with those estimated over ten years earlier with the simpler STCP code and an earlier version of MELCOR—and for different plants—is expected to show differences. A more logical and meaningful approach is to compare contemporary, severe accident computer code models and compare their predictions to experimental data or to postulated reactor conditions for the same scenario. In other words, the best comparison is of the predictions of computer models at the same point in time with the same inputs and available data available to the code analysts.

65. The NRC, the nuclear power industry, and the international nuclear energy research community have extensively researched and studied accident phenomena and offsite consequences of severe reactor accidents over the last 25 years. As part of an initiative to assess plant response to security-related events following the terrorist attacks of 2001, the NRC completed updated analyses of severe accident progression and offsite consequences. Those analyses incorporate a wealth of accumulated research data as well as more detailed, integrated, and realistic modeling methods than previous analyses. One insight gained from these security assessments was that updated analyses of severe reactor accidents were needed to reflect realistic estimates of the more likely accident outcomes given the current state of plant design and operation and advances in our understanding of severe accident behavior. NUREG-1935, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report, Draft Report for Public Comment" at xv (Jan. 2012) ("Draft NUREG-1935") (Attach. 39).

66. Consequently, the NRC initiated the State-of-the-Art Reactor Consequence Analyses ("SOARCA") project in 2006 to develop revised best estimates of the offsite

radiological health effect consequences of severe reactor accidents. The project's principal objective was to develop updated and more realistic severe accident analyses by including significant plant changes and reactor safety research updates not reflected in earlier NRC assessments. SOARCA included consideration of plant system improvements, improvements in training and emergency procedures, offsite emergency response, and security-related improvements, as well as plant changes such as power uprates and lengthened operating times.

67. The SOARCA analyzed two plants that are typical of the two U.S. commercial reactor types, *i.e.*, a BWR plant, the Peach Bottom Atomic Power Station in Pennsylvania, and a PWR plant, Surry Power Station in Virginia. These two plants also took part in earlier accident analyses performed by the NRC, including the seminal WASH-1400 PRA study (1975), the Sandia Siting Study (1982),[18] and the NUREG-1150 (1990) study. The SOARCA analysis considered one plant unit at each site.

68. The SOARCA project used computer-modeling techniques to understand how a reactor might behave under severe accident conditions, and how a release of radioactive material from the plant might affect the public. Specifically, it used MELCOR to model the severe accident scenarios within the plant and MACCS2 to model the offsite health effect consequences of any atmospheric releases of radioactive material.

69. In January 2012, the NRC published the results of its assessment and plant-specific reports for Peach Bottom and Surry. *See* Draft NUREG-1935; NUREG/CR-7110, "State-of-the-Art Reactor Consequence Analyses Project Volume 1: Peach Bottom Integrated Analysis" (Jan. 2012) (Attach. 41); NUREG/CR-7110, "State-of-the-Art Reactor Consequence Analyses Project Volume 2: Surry Integrated Analysis" (Jan. 2012) (Attach. 42). Among the findings, the NRC found that, in addition to delayed radiological releases, the magnitude of the radionuclide release,

[18] NUREG/CR-2239, "Technical Guidance for Siting Criteria Development" (1982) (excerpt attached as Attach. 40).

especially with respect to the key radionuclide (iodine and cesium) groups, is much smaller than estimated in prior studies, such as the 1982 Sandia Siting Study. This is shown in Figure 4 below.

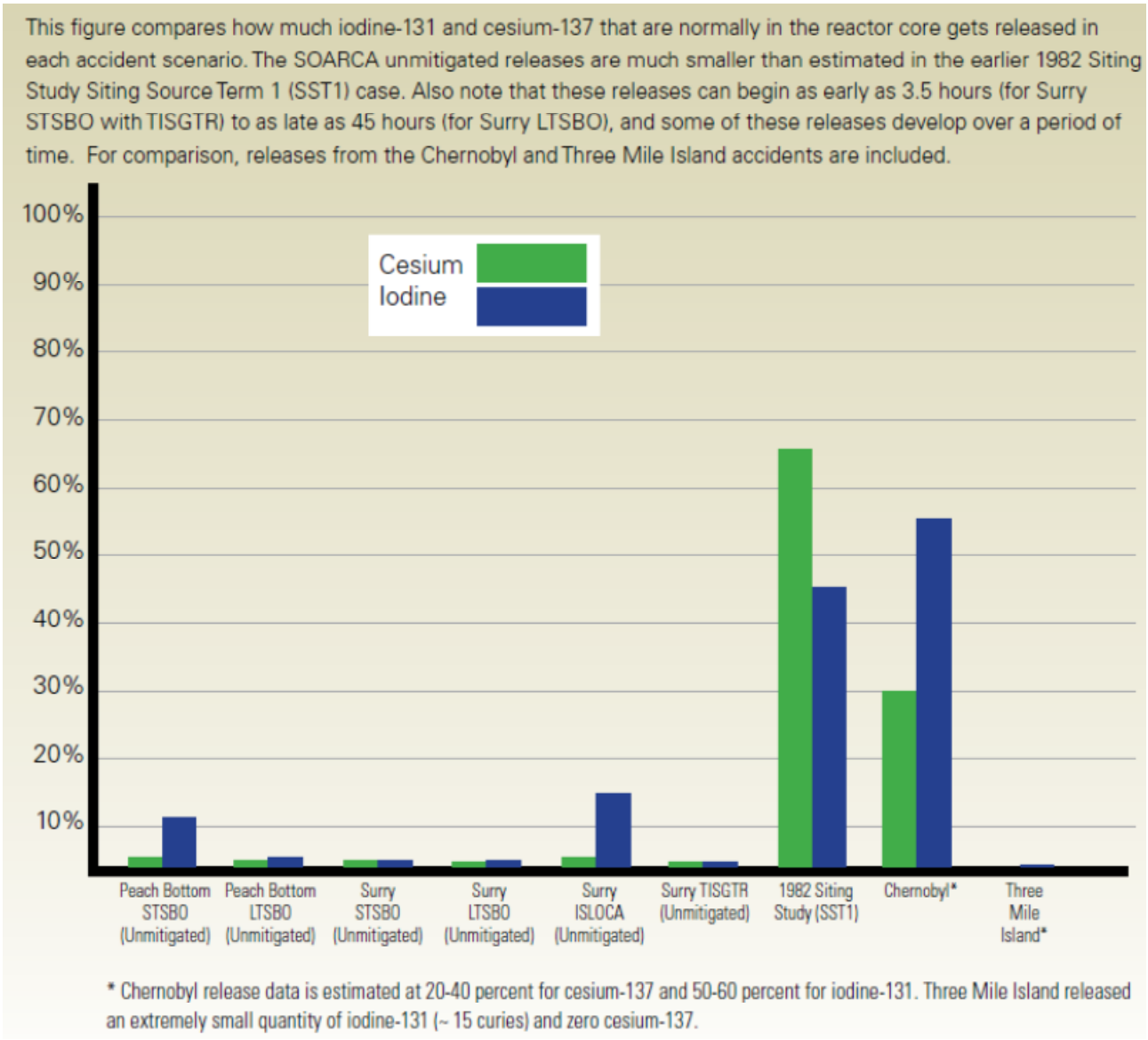

Figure 4. Percentages of Iodine and Cesium Released to the Environment During the First 48 Hours of the Accident for SOARCA Unmitigated Scenarios, 1982 Siting Study (SST1), and Historical Accidents (from NUREG/BR-0359, "Modeling Potential Reactor Accident Consequences," 24 fig. 4.1 (Jan. 2012) (Attach. 43).

70. Another more meaningful comparison is between the MAAP and MELCOR codes. Sandia developed MELCOR for the NRC, and it is the current software tool in a line of evolutionary, severe accident progression computer models used by the NRC. As noted above,

Sandia and the NRC used MELCOR in support of the SOARCA project. Both MAAP and MELCOR are now used throughout the world and are much more advanced than predecessor versions or simpler models, such as the STCP, that were applied more than 20 years ago in the NUREG-1150 study. They are both integrated codes that allow the calculation of accident sequences from the initiating event while taking into account important inter-related phenomena (*e.g.*, reactor coolant system, containment thermal-hydraulics, in-vessel core degradation, molten core concrete interaction, fission product release and transport into the environment).

71. A 2004 comparison using MELCOR and MAAP for a PWR accident sequence demonstrates that the two codes provide similar calculated results for thermal-hydraulic and core degradation response of the plant, with minor differences in various timings of phenomena. The authors indicated that these minor differences in results were within the uncertainties of the code numerical computations and the physics models. K. Vierow, Y. Liao, J. Johnson, M. Kenton, and R. Gauntt, "Severe accident analysis of a PWR station blackout with the MELCOR, MAAP4, and SCDAP/RELAP5 Codes," *Nuclear Engineering and Design* 234, 129-145 (2004) (Attach. 44). Although this documented comparison did not specifically address the calculation of release fractions to the environment, this comparison does support the use of either code for purposes supporting PRAs. In our judgment, the results of this comparison show that use of MAAP is reasonable for the purposes of developing a SAMA analysis for the purposes of NEPA.

72. Notably, both MAAP and MELCOR were used soon after the March 2011 Fukushima Dai-ichi nuclear power plant accident in Japan. Tokyo Electric Power Company, the operating utility for the six-unit station, has used MAAP to inform its understanding of the accident progression in Units 1-3 during the earthquake and subsequent tsunami event in March 2011. International Atomic Energy Agency, *IAEA International Fact Finding Expert Mission of the Fukushima Dai-ichi NPP Accident Following the Great East Japan Earthquake and Tsunami*,

at 33-35 (June 2011) (Attach. 45). Sandia applied MELCOR in modeling the Station Blackout sequence for the NRC in support of the Japanese Government.

73. The contemporary applications and comparisons of MAAP discussed above demonstrate the current-day use and value of MAAP (and MELCOR) with respect to state-of-the-art modeling and simulation of severe reactor accident conditions. In our professional opinions, they are better indicators of the MAAP code's fitness for simulating severe accident conditions and estimating environmental source terms than the references cited by Intervenors, which are outdated and impertinent and which fail to show any flaw in the MAAP code or related inputs used by FirstEnergy in its SAMA analysis

VI. CONCLUSION

74. We have thoroughly evaluated the claims in Contention 4 against information in the recently amended Davis-Besse ER and its supporting technical documentation, the applicable accepted standards for performing PRAs and SAMA analyses, and the studies and reports discussed above. Based on our evaluation, we conclude that all of the Intervenors' claims lack a technical foundation and provide no reason to conclude that the source terms used in the Davis-Besse SAMA analysis are invalid or unreasonable. Specifically we conclude the following:

- The MAAP code has a strong technical basis for use in PRA and severe accident analysis and has been accepted for use in numerous NRC-approved analyses. Use of the MAAP code is reasonable for a SAMA analysis performed under NEPA.
- The use of plant-specific source terms (*e.g.*, based on MAAP) is preferred over the use of generic source terms (*e.g.*, based on NUREG-1465) for a SAMA analysis where plant specific design and operational changes are evaluated.
- The primary purpose of NUREG-1465 source terms is for defining releases into containment, not to the environment. A SAMA analysis requires a plant-specific evaluation of releases to the environment.
- NUREG-1465 provides data only for a single PWR release. A SAMA analysis requires an evaluation of the spectrum of plant-specific releases. Use of NUREG-1465 data for the entire spectrum would result in grossly-distorted SAMA results.

In accordance with 28 U.S.C. § 1746, we declare under penalty of perjury that the foregoing is true and correct.

Executed in accord with 10 C.F.R. § 2.304(d)
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July 26, 2012

Attachment A

Definitions of Key Severe Accident and PRA Terms

Term	Definition
Accident Progression Bin	A group of postulated accidents that has similar characteristics with respect to the timing of containment building failure and other factors that determine the amount of radioactive material released. Accident progression bins are sometimes referred to as containment failure modes in older PRAs.
Core Damage Frequency	The frequency of combinations of initiating events, hardware failures, and human errors leading to core uncovery with reflooding of the core not imminently expected.
Core Inventory	The amount (in units of activity) of each radionuclide present in the reactor core at the time accident initiation.
External Initiating Events	Events occurring away from the reactor site that result in initiating events in the plant. In keeping with PRA tradition, some events occurring within the plant during normal power plant operation, *e.g.*, fires and floods initiated within the plant, are included in this category.
Initiating Event	A challenge to plant operation from which there can be numerous accident sequences. The various accident sequences result regardless of whether plant systems operate properly or fail and what actions operators take. Some accident sequences will result in a safe recovery and some will result in reactor core damage. PRA normally consider internal and external initiating events.
Internal Initiating Events	Initiating events involving components internal to the plant (*e.g.*, transient events requiring reactor shutdown, pipe breaks) occurring during the normal power generation of a nuclear power plant. In keeping with PRA standard practice, loss of offsite power is considered an internal initiating event.
Plant Damage State	A group of accident sequences that has similar characteristics with respect to accident progression and containment engineered safety feature operability.
Release Fraction	The fraction defining the portion of the radionuclide inventory by radionuclide group in the reactor at the start of an accident that is released through a containment barrier(s), such as the reactor coolant system, to the primary containment, or from the primary containment to the environment.
Severe Accident	Severe nuclear accidents are those in which substantial damage is done to the reactor core whether or not there are serious offsite consequences. A severe accident is often described as a beyond design-basis accident involving multiple failures of equipment or function. Although severe accidents generally have lower likelihoods than design-basis accidents, they may have greater consequences.
Source Term	The fractions of the core inventory released to the atmosphere, and the timing and other release information needed to calculate the offsite consequences. Specifically, the information includes the fractions of the radionuclide groups in the inventory in the reactor at the start of an accident that are released to the containment (*i.e.*, the containment source term), or to the environment (*i.e.*, the environmental source term). The source term to the environment also includes the initial elevation, heat or energy content of the plume, and timing of the release (time after accident initiation or shutdown, and duration of release).

ATTACHMENT 3

Curriculum Vitae of Dr. Kevin R. O'Kula

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KEY AREAS:

- **Computer Model Verification and Validation**
- **Accident and Consequence Analysis for Design Basis Accident Support**
- **Regulatory Standard & Guidance Development**
- **New Reactor Design Accident Analysis and PRA Support**
- **Severe Accident and Quantitative Risk Analysis**
- **Level 2/3 Probabilistic Risk Assessment**
- **Tritium Dispersion and Consequence Analysis**
- **MACCS2 Code Applications**
- **Level 3 PRA Standard Development**

PROFESSIONAL SUMMARY:

Dr. O'Kula has over 29 years of experience as a manager and technical professional in the areas of accident and consequence analysis, source term evaluation, commercial and production reactor probabilistic risk assessment (PRA) and severe accident analysis, safety software quality assurance (SQA), safety analysis standard and guidance development, computer code evaluation and verification, risk management, hydrogen safety, reactor materials dosimetry, shielding, and tritium safety applications. Kevin is currently the lead for the PRA technical area in the Risk Assessment and Analysis Group. He is a member of the American Nuclear Society (ANS) Standard working group ANS 58.25 on Level 3 Probabilistic Safety Assessment, and recently concluded activities as a member of the Peer Review Committee for the Nuclear Regulatory Commission's (NRC's) State-of-the-Art Reactor Consequence Analysis (SOARCA) Program. Dr. O'Kula was part of the Department of Energy (DOE) team writing DOE G 414.1-4, *Safety Software Guide*. He coordinated technical support for the DOE Office of Environment, Safety, and Health (EH) in addressing Defense Nuclear Facilities Safety Board (DNFSB) Recommendation 2002-1 on Software Quality Assurance (SQA), and was a consultant to DOE/EH-31 Office of Quality Assurance for disposition of SQA issues.

Dr. O'Kula is supporting, or has supported, Atomic Safety Licensing Board (ASLB) relicensing issue resolution for several commercial nuclear power plants, including Indian Point Units 2 and 3, Davis-Besse Nuclear Power Station, Prairie Island Units 1 and 2, and Pilgrim Nuclear Power Station, on severe accident mitigation alternatives (SAMA) analysis issues. He also was part of the accident analysis and PRA/severe accident teams supporting the Design Certification Document for the U.S. Advanced Pressure Water Reactor (US-APWR) a joint effort with URS Washington Division and Mitsubishi Heavy Industries (MHI). He has provided similar support for an alternative reactor technology, the Pebble Bed Modular Reactor (PBMR).

Kevin was a member of the Partner, Assess, Innovate, and Sustain (PAIS) Safety Case team for the Sellafield Sites in the United Kingdom in the early 2009 period. The PAIS team identified and began implementation of improvement opportunities in nuclear safety and related areas for Sellafield. Recommendations were documented in comprehensive reports to the Site's Nuclear Management Partners consortium in March 2009.

Dr. O'Kula is coordinating URS SMS support to the Quantitative Risk Analysis (QRA) for evaluation of hydrogen events to risk-inform the Waste Immobilization and Treatment Plant (WTP) design at Hanford, including fault tree analysis and reliability data, and human factors areas. He is also a contributor to the DOE response on the use of risk assessment methodologies as part of the DNFSB Recommendation 2009-1 implementation action for Risk Assessment. He led work in reviewing EIS food pathway consequence analysis performed on assumed accident conditions from the Mixed Oxide Fuel Fabrication Facility (MFFF), sited at the Savannah River Site. This project compared and evaluated the impacts calculated from three computer models, including MACCS2, GENII, and UFOTRI.

Kevin is past chair of the ANS Nuclear Installations Safety Division (NISD), and the Energy Facility Contractors Group (EFCOG) Accident Analysis Subgroup, and is the current NISD Program Committee Chair. He is a member of the Nuclear Hydrogen Production Technical Group under the ANS's Environmental Sciences Division, and is chair for the EFOCG Hydrogen Safety Interest Group. He was the Technical Program Chair for two ANS embedded topical meetings on Operating Nuclear Facility Safety (Washington, D.C., 2004) and the Safety and Technology of Nuclear Hydrogen Production, Control and Management (Boston, MA, 2007). He is the Assistant Technical Program Committee Chair for the Probabilistic Safety Assessment (PSA) Meeting in Columbia, SC, scheduled for September 22-26, 2013.

Dr. O'Kula was PRA group manager for K Reactor at the time of restart in the early 1990s. He led a successful effort demonstrating Savannah River Site (SRS) K-Reactor siting compliance to 10 CFR Part 100, and tritium facility compliance with SEN-35-91.

Kevin was the project leader for independent Verification and Validation (V&V) of urban dispersion software for the Defense Threat Reduction Agency (DTRA) and is the current V&V project manager for the evaluation of several chemical/biological software tools for the U.S. Army Test and Evaluation Command (ATEC) and Chemical-Biological Program (Dugway Proving Ground (Utah) and Edgewood Chemical/Biological Center in Maryland.

EDUCATION:

Ph.D., Nuclear Engineering, University of Wisconsin, 1984
M.S., Nuclear Engineering, University of Wisconsin, 1977
B.S., Applied and Engineering Physics, Cornell University, 1975

TRAINING:

Conduct of Operations (CONOPS), 1994
Harvard School of Public Health, Atmospheric Science and Radioactivity Releases, 1995
Consequence Assessment, (Savannah River Site, 1995)
U.S. DOE Risk Assessment Workshop (Augusta, GA, 1996)
MELCOR Accident Computer Code System (MACCS) 2 Computer Code, 1997, 2005, 2011
MCNPX Training Class (ANS Meeting, 1999)

CLEARANCE:

Active DOE "L"

PROFESSIONAL EXPERIENCE:

URS Safety Management Solutions LLC **1997 to Present**
Advisory Engineer and Senior Fellow Advisor

Dr. O'Kula recently concluded activities as a member of the NRC-Sandia National Laboratories State-of-the-Art Reactor Consequence Analysis (SOARCA) Project Peer Review Committee. The SOARCA team provided recommendations on applying MACCS2 for modeling accident phenomena and subsequent off-site consequences from postulated severe reactor accidents. This activity supports the efforts of Sandia National Laboratories (SNL) and the Nuclear Regulatory Commission (NRC) to provide more realistic assessment of severe accidents.

Kevin also part of the Level 3 PRA Standard working group charged with developing an ANSI/ANS standard for Level 3 PRA analysis. He participated in a team that conducted an SQA gap analysis on the bioassay code [Integrated Modules for Bioassay Analysis (IMBA)] based on DOE G 414.1-4 requirements. He identified safety analysis codes that were designated as DOE "toolbox" codes, and oversaw production of the first documents (QA criteria and application plan, code guidance reports, and gap analysis) for six accident analysis codes designated for the DOE Safety Software Toolbox. He provided support to DOE/EH-31 (now DOE/HSS) for addressing SQA issues for safety analysis software. He was a contributor to DOE G 414.1-4, *Safety Software Guide* on SQA practices, procedures, and programs.

Kevin has provided technical input for work packages on several recent commercial projects. In the first, he teamed with Entergy on MACCS2 code applications issues in the Severe Accident Mitigation Alternatives (SAMA) analysis area for the Pilgrim Nuclear Power Station. In the second, he was part of tritium environmental release analysis team that supported evaluation of tritium control and management areas for the Braidwood plant. A third effort developed an initial SAMDA document for the Mitsubishi Heavy Industries (MHI) US-APWR (1610 MW_e evolutionary PWR), as well as complete a control room habitability study for postulated toxic chemical gas releases.

He was part of a Washington Group team that developed a Design Control Document (DCD) for the MHI US-APWR using input information from MHI. He was Chapter lead on Chapter 15 (Transient and Accident Analysis), and later transitioned to severe accident evaluation and documentation support to Chapter 19 (PRA and Severe Accidents). He was the Chapter 19 lead for PRA and Severe Accident for COLA development for the Pebble Bed Modular Reactor (PBMR).

Dr. O'Kula developed the outline, coordinated contributors, and assembled the first draft of the DOE *Accident Analysis Guidebook*, a reference guide for hazard, accident, and risk analysis of nuclear and chemical facilities operated in the DOE Complex. He is also the primary author and coordinator for the *Accident Analysis Application Guide* for the Oak Ridge contractor. Dr. O'Kula also developed a one-day course and exam for the guide, which he later presented to the Oak Ridge, Paducah, and Portsmouth staff.

Kevin also led an independent V&V review for the DTRA of the U.K.-developed Urban Dispersion Model (UDM) software for predicting chemical and biological plume dispersion in city environments, and is leading projects to verify and validate chemical/biological simulation suite software applications for the Dugway Proving Ground (Utah), and the Edgewood Chemical Biological Center (ECBC) in Maryland.

Managing Member, Consequence Analysis

Dr. O'Kula was responsible for the consequence analysis associated with accident analysis sections of Documented Safety Analysis (DSA) reports and other safety basis documents for SRS, Oak Ridge, and other DOE nuclear facilities. He also developed the methodology and identified appropriate computer models for this purpose. Additionally, Dr. O'Kula developed training to enhance consistency and

standardize analyses in the consequence analysis area. He was project manager for environmental assessment support to SRS on a transportation safety analysis using the RADTRAN code.

Kevin coordinated development of a DOE Accident Analysis Guidebook involving over 10 sites and organizations. He also led the effort to produce Computer Model Recommendations for source term (fire, spill, and explosion), in-facility transport, and dispersion/consequence (radiological and chemical) areas.

Westinghouse Savannah River Company **1989 to 1997**
Group Manager

Dr. O'Kula managed consequence analyses associated with accident analysis sections of DSA reports and other safety basis documents. He also developed the associated methodologies and identified appropriate computer models. He was a member of the management team supporting Criticality Safety Evaluation preparation assisting Safe Sites of Colorado and the dispositioning of final criticality safety issues for the decommissioning and decontamination of nuclear facilities at the Rocky Flats Environmental Technology Site.

In a teaming arrangement with Science Applications International Corporation, Kevin initiated discussions that led to development of an emergency management enhancement tool to risk inform likely source terms. He applied this approach to a Savannah River nuclear facility (K Reactor), and was part of the team to provide this methodology for use on the British Advanced Gas-Cooled Reactors (AGRs) (for the United Kingdom's Nuclear Installation Inspectorate). The model was knowledge-based and required the development of an Accident Progression Event Tree (APET) for the facility in question.

Dr. O'Kula managed the completion of the SRS K Reactor PRA program. He was the lead for development of the K Reactor Source Term Predictor Model and assisted with the core technology lay-up program to preserve competencies in reactor safety. He coordinated a 25-person group responsible for K Reactor probabilistic and deterministic dose analyses, and led the examination of reduced power cases at project termination. He developed risk and dose management applications to cost-effectively prioritize facility modifications.

Kevin interfaced with DOE Independent and Senior Review teams to finalize study acceptance, and transitioned the risk assessment team to risk management functions for nuclear and waste processing facilities. In addition, he successfully prepared a 10 CFR 100 Siting white paper to resolve issues raised by the DNFSB, and teamed with DOE/HQ legal support to document resolutions. He led the development of a position paper demonstrating SRS Replacement Tritium Facility compliance with DOE Safety Policy (SEN-35-91).

Staff Engineer

Dr. O'Kula led an analytical team quantifying the tritium source term during a Loss of River Water design basis accident. He evaluated airborne tritium levels with multi-cell CONTAIN model, interfaced with a multidisciplinary team to resolve Operational Readiness Review concerns, developed an SRS-specific methodology for applying MACCS as a tool for Level 3 PRA Applications, and applied CONTAIN code for K Reactor source term analysis.

E.I. du Pont de Nemours & Company **1982 to 1989**
Principal Engineer, Research Engineer

Dr. O'Kula performed risk analysis duties for the Savannah River Laboratory (SRL) Risk Analysis Group, after earlier conducting research activities for the Reactor Materials and Reactor Physics Groups. He performed initial planning for offsite irradiation of test specimens to evaluate remaining reactor lifetime for Savannah River reactor components.

Westinghouse Electric Corporation **1975**
Summer Student, Reactor Licensing
Monroeville, PA

American Electric Power Corporation **1973 to 1974**
Co-op Student, Reactor Physics and Reactor Licensing
New York, NY

Long Island Lighting Company **1972**
Summer Intern
Riverhead, NY

PARTIAL LIST OF PUBLICATIONS (2000-2011):

M. G. Wentink, K. R. O'Kula (Primary and Presenting Author), H. A. Ford, C.R. Lux, and H. C. Benhardt, *Operational Frequency Analysis Model Supporting the QRA for Risk-Informing the Design of a Waste Processing Facility***,** American Nuclear Society Winter Meeting, October 30 - November 3, 2011 (Washington, D.C.).

K. R. O'Kula, D. C. Thoman, J. Lowrie, and A. Keller, *Perspectives on DOE Consequence Inputs for Accident Analysis Applications*, American Nuclear Society 2008 Winter Meeting and Nuclear Technology Expo, November 9-13, 2008 (Reno, NV).

K. R. O'Kula, F. J. Mogolesko, K-J Hong, and P. A. Gaukler, *Severe Accident Mitigation Alternative Analysis Insights Using the MACCS2 Code*, American Nuclear Society 2008 Probabilistic Safety Assessment (PSA) Topical Meeting, September 7-11, 2008 (Knoxville, TN).

K. R. O'Kula and D. C. Thoman, *Modeling Atmospheric Releases of Tritium from Nuclear Installations*, American Nuclear Society Embedded Topical Meeting on the Safety and Technology of Nuclear Hydrogen Production, Control and Management, June 24-28, 2007 (Boston, MA).

K. R. O'Kula and D. C. Thoman, *Analytical Evaluation of Surface Roughness Length at a Large DOE Site (U)*, American Nuclear Society Winter Meeting, November 12-16, 2006 (Albuquerque, NM).

K. R. O'Kula and D. Sparkman, *Safety Software Guide Perspectives for the Design of New Nuclear Facilities (U),* Winter Meeting of the American Nuclear Society, November 13 – 17, 2005 (Washington, D.C.).

K. R. O'Kula and R. Lagdon, *Progress in Addressing DNFSB Recommendation 2002-1 Issues: Improving Accident Analysis Software Applications*, Fifteenth Annual Energy Facility Contractors Group Safety Analysis Workshop, April 30 – May 5, 2005, Los Alamos, NM (2005).

K. R. O'Kula and Tony Eng, *A "Toolbox" Equivalent Process for Safety Analysis Software*, Fourteenth Annual Energy Facility Contractors Group Safety Analysis Workshop, May 1-6, 2004, Pleasanton, CA (2004).

K. R. O'Kula, D. C. Thoman, J. A. Spear, R. L. Geddes, *Assessing Consequences Due to Hypothetical Accident Releases from New Plutonium Facilities (U)*, American Nuclear Society Embedded Topical Meeting on Operating Nuclear Facility Safety, November 14 – 18, 2004 (Washington, D.C.).

K. O'Kula and J. Hansen, *Implementation of Methodology for Final Hazard Categorization of a DOE Nuclear Facility (U)*, Annual Meeting of the American Nuclear Society, June 13-17, 2004,

(Pittsburgh, PA).

K. R. O'Kula and Tony Eng, *A "Toolbox" Equivalent Process for Safety Analysis Software*, Fourteenth Annual Energy Facility Contractors Group Safety Analysis Workshop, May 1-6, 2004, Pleasanton, CA (2004).

K. R. O'Kula, et al., *Evaluation of Current Computer Models Applied in the DOE Complex for SAR Analysis of Radiological Dispersion & Consequences*, WSRC-TR-96-0126, Westinghouse Savannah River Company (2003).

K. R. O'Kula, et al., *Evaluation of Current Computer Models Applied in the DOE Complex for SAR Analysis of Radiological Dispersion & Consequences*, WSRC-TR-96-0126, Rev. 3, Westinghouse Savannah River Company (2002).

K. R. O'Kula, *A DOE Computer Code Toolbox: Issues and Opportunities*, Eleventh Annual EFCOG Workshop, also 2001 Annual Meeting of the American Nuclear Society, Milwaukee, WI (2001).

PUBLICATIONS (1988-1999):

Dr. O'Kula authored or co-authored more than 20 publications between 1988 and 1999. Details are available upon request.

PROFESSIONAL SOCIETIES AND STANDARDS COMMITTEES

- American Nuclear Society
- Health Physics Society
- ANS Level 3 PRA Standard Committee 58.25

ATTACHMENT 4

Curriculum Vitae of Mr. Grant A. Teagarden

GRANT TEAGARDEN

Manager
Consequence Analysis

AREAS OF EXPERTISE

- ***Level 3 PRA (MACCS)***
- ***Security Risk Assessment***
- ***Level 2 PRA (MAAP)***
- ***Fire PRA***
- ***Internal Flooding***
- ***Data Analysis***

EDUCATION

B.S., Mechanical Engineering, University of Miami, Florida

Bettis Nuclear Reactor Engineering School, Bettis Atomic Power Laboratory, Pennsylvania

SECURITY CLEARANCE

Secret

Safeguards

U.S. Citizen

WORK EXPERIENCE SUMMARY

Mr. Teagarden has over fourteen years of experience in the nuclear field, including four years as a Naval Reactors Engineer in the U.S. Navy. He is experienced in Level 3 PRA consequence analysis, Fire PRA, plant security risk assessment, PRA updates, data and common cause failure analysis, integrated leak rate test extension evaluations, and internal flood updates.

WORK EXPERIENCE

Mr. Teagarden holds a Bachelor of Science degree in Mechanical Engineering from the University of Miami, Florida. He is ERIN Engineering's lead for Level 3 PRA consequence analysis (e.g. radiological dispersion analysis). The following are some of his recent consequence analysis activities:

- Developed Level 3 PRA models (MACCS2) for Limerick (2011), Callaway (2011), Diablo Canyon (2009), Salem (2008), Hope Creek (2008), Progress Energy Levy County Site (2008), Harris Advanced Reactor Site (2007), General Electric's ABWR (2007) and ESBWR (2006, 2005), TMI (2006), Prairie Island (2006), Oyster Creek (2004), Exelon Early Site Permit (2004), Palisades (2004), and Monticello (2004)
- Supported Level 3 PRA SAMA contention resolution for Davis Besse (2011), Indian Point (2009-2011) and Prairie Island (2008)
- Developed quasi-site specific Level 3 PRA models (MACCS2) for every operating U.S. commercial nuclear power plant site in support of industry security assessments (2005)
- Vice Chair of ANS Level 3 PRA Standard Writing Committee (ANS-58.25)

Mr. Teagarden has been an integral part of ERIN Engineering's security assessment team, developing and implementing risk based assessment methodologies for the commercial nuclear power plant industry in the U.S. and clients abroad. The following are some of his recent activities:

- Supported Aircraft Impact Analysis per NEI 07-13 guidance for GE ABWR (2009-2010), KOPEC APR-1400 (2010), MHI US-APWR (2007-2010), and MHI EU-APWR (2009)
- Co-authored development of Risk Analysis and Management for Critical Asset Protection (RAMCAP) methodology for nuclear power plants in support of EPRI, ASME, and the U.S. Department of Homeland Security (DHS) and supported its implementation for NEI at all U.S. nuclear power plants (2005-2007). Implementation included leading an NEI sponsored industry workshop on the methodology, coordinating and facilitating site assessments, and developing industry level insights from the assessment results.
- Developed RAMCAP methodology for spent nuclear fuel dry storage and transportation in support of EPRI, ASME, and DHS, and supported its implementation for NEI (2005-2007)

PROFESSIONAL ORGANIZATIONS

American Nuclear Society

- Provided RAMCAP training for personnel at Idaho National Laboratory
- Co-authored report for EPRI for identifying potential mitigation strategies for beyond design basis conditions (2005)
- Participated in the development of NEI industry guidance to resolve security related open issues related to large fires and explosions (i.e., B.5.b) at all U.S. nuclear power plants (2005-2006)
- Authored Vulnerability Assessment Methodology for EPRI and NEI use for security threat analysis (2003)
- Co-authored report for EPRI for identification of mitigation strategies for scenarios involving loss of intake structure and offsite power (2004)
- Co-authored reports for EPRI and NEI use for operational response to beyond design basis security threats (2003)
- Authored Explosive Threat Guidelines for EPRI and NEI use in response to NRC Interim Compensatory Measures (2002)

Mr. Teagarden is experienced with Level 2 PRA (e.g., severe accident analysis) and the use of the thermal hydraulic MAAP code to model severe accident phenomenology. Some of his recent Level 2 PRA activities include:

- Managed updates of Hatch Level 2 PRA (2009-2010)
- Performed Level 2 MAAP runs for Columbia Generating Station (2010) in support of SAMA life extension
- Supported Level 2 PRA updates and MAAP analysis for Grand Gulf (2010) and Limerick (2010)
- Served as analyst for Integrated Leak Rate Test (ILRT) extension evaluations for Hatch (2010), Clinton (2006), Columbia (2004), Dresden (2003) and Quad Cities (2002)
- Performed independent review for ILRT extension evaluation for NMP1 and NMP2 (2008, 2009)

Mr. Teagarden has been involved with the commercial nuclear industry's development of Fire PRAs using the guidance of NUREG/CR-6850 in the following ways:

- Exelon Fire PRA Model Owner for Clinton Power Station (2011)
- Technical lead for Fire PRA project for KKM (2011)
- Supported Fire PRA updates to NUREG/CR-6850 for Hope Creek (2010), LaSalle (2008-2009), Clinton (2007-2008) and Hatch (2007)

Mr. Teagarden has also been involved in other technical aspects of PRA analyses such as:

- Served as analyst for Hope Creek and Quad Cities PRA Updates performing the common cause failure analysis and revisions to all the System Notebooks (2002)
- Served as analyst for NASA Space Shuttle PRA performing common cause failure analysis (2002)

- Performed pipe rupture and flooding analysis for Internal Flood updates for South Texas Project Units 3&4 (2010), Clinton (2009), Hope Creek (2003), Dresden (2001), and Oyster Creek (2001)
- Performed system analysis and pipe rupture evaluation for Limerick ISLOCA analysis (2001)

Prior to working for ERIN Engineering, Mr. Teagarden worked four years as a Naval Reactors Engineer in the U.S. Navy for the Department of Energy. He was principally involved in refueling operations, providing technical support and oversight for the nuclear refueling of the eight reactors on the USS Enterprise Aircraft Carrier. Mr. Teagarden's responsibilities included oversight of reactor disassembly, spent fuel removal and shipout, and reactor reassembly.

ATTACHMENT 5

Letter from B. Allen, FirstEnergy, to NRC Document Control Desk, "Correction of Errors in the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4613) Environmental Report Severe Accident Mitigation Alternatives Analysis, and License Renewal Application Amendment No. 29" (July 16, 2012)

FENOC
FirstEnergy Nuclear Operating Company

Davis-Besse Nuclear Power Station
5501 N. State Route 2
Oak Harbor, Ohio 43449

July 16, 2012
L-12-244

10 CFR 54

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Davis-Besse Nuclear Power Station, Unit No. 1
Docket No. 50-346, License Number NPF-3
Correction of Errors in the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4613) Environmental Report Severe Accident Mitigation Alternatives Analysis, and License Renewal Application Amendment No. 29

By letter dated August 27, 2010, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102450565), FirstEnergy Nuclear Operating Company (FENOC) submitted an application pursuant to Title 10 of the *Code of Federal Regulations*, Part 54 for renewal of Operating License NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse). In January 2012, during review of the License Renewal Application (LRA), Environmental Report, Attachment E, Severe Accident Mitigation Alternatives (SAMA) Analysis, four errors were identified that affected the SAMA Analysis. In March 2012, during review of the draft corrected SAMA Analysis and an extent-of-condition review for the four errors, an additional error was identified in the analysis. On January 12 and March 28, 2012, FENOC contacted Ms. Paula Cooper, Nuclear Regulatory Commission (NRC) Environmental Project Manager, to inform NRC of the SAMA Analysis errors and discuss the impacts to the SAMA Analysis review schedule. Following correction of the five errors, the revised (corrected) SAMA Analysis conclusions did not change; specifically, the revised SAMA Analysis did not result in the discovery of any additional cost beneficial SAMAs beyond the one (SAMA AC/DC-03, which adds a portable diesel-driven battery charger to the DC system) identified in the FENOC letter dated June 24, 2011 (ML11180A233).

Attachment 1 provides a description of the five SAMA Analysis errors.

Attachment 2 provides, based on the revised SAMA Analysis, the results of a review for impacts to the responses to NRC requests for additional information (RAIs) for the SAMA Analysis submitted by FENOC letter dated June 24, 2011 (ML11180A233).

Attachment 3 provides, based on the revised SAMA Analysis, the results of a review for impacts to the supplemental responses to NRC supplemental RAIs for the SAMA Analysis submitted by FENOC letter dated September 1, 2011 (ML11250A068).

The Enclosure provides Amendment No. 29 to the Davis-Besse LRA.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Clifford I. Custer, Fleet License Renewal Project Manager, at 724-682-7139.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 16, 2012.

Sincerely,

John C. Dominy
Director, Site Maintenance

Attachments:
1. Description of Errors Identified in the Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse), License Renewal Application, Environmental Report, Attachment E, Severe Accident Mitigation Alternatives Analysis
2. Review for Impacts to Responses to Requests for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse), License Renewal Application, Environmental Report, Attachment E, Severe Accident Mitigation Alternatives Analysis Submitted by FENOC Letter Dated June 24, 2011 (ML11180A233)
3. Review for Impacts to Responses to Requests for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse), License Renewal Application, Environmental Report, Attachment E, Severe Accident Mitigation Alternatives Analysis Submitted by FENOC Letter Dated September 1, 2011 (ML11250A068)

Enclosure:
Amendment No. 29 to the Davis-Besse License Renewal Application

cc: NRC DLR Project Manager
NRC DLR Environmental Project Manager
NRC Region III Administrator

cc: w/o Attachments or Enclosure
NRC DLR Director
NRR DORL Project Manager
NRC Resident Inspector
Utility Radiological Safety Board

Attachment 1
L-12-244

Description of Errors Identified in the Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse), License Renewal Application, Environmental Report, Attachment E, Severe Accident Mitigation Alternatives Analysis

During reviews of the Severe Accident Mitigation Alternatives (SAMA) Analysis, the following five errors were identified:

1. An inaccurate land area conversion factor for acres to hectares was used.
2. Dollar values for Ohio farmland and non-farmland used as inputs to the "MELCOR Accident Consequence Code System" (MACCS2) software used in support of the SAMA Analysis were not appropriate. The land values were selected from Ohio Department of Taxation 'tax assessment' values instead of 'appraised' values. The Ohio tax assessment value is 35 percent of the appraised value.
3. The escalation of decontamination costs used in the SAMA Analysis was not performed per the guidance of Nuclear Energy Institute (NEI) 05-01 "Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document," November 2005, using the consumer price index.
4. Use of core inventory isotopic 'activity' instead of isotopic 'mass' in the Modular Accident Analysis Program (MAAP) software code runs did not reflect updated industry guidance. MAAP Users Group News Bulletin, "MAAP-FLASH #68" (August 5, 2008), recommended that users of MAAP versions 4.0.5 through 4.0.7 (FENOC is currently using MAAP software version 4.0.6) include plant-specific values for the mass of the relevant fission product elements instead of the isotopic activity of those elements.
5. The wind direction from the Davis-Besse Meteorological Tower was not converted from the 'blowing from' direction to the 'blowing toward' direction for use in the SAMA Analysis calculations. The data from the Davis-Besse Meteorological Tower is received in the 'blowing from' direction. However, the MACCS2 software requires wind direction data inputs to be provided in the 'blowing toward' direction. The data conversion was not performed properly.

Following correction of the five errors identified above, the revised (corrected) SAMA Analysis conclusions did not change; specifically, the revised SAMA Analysis did not result in the discovery of any additional cost-beneficial SAMAs beyond the one (SAMA AC/DC-03, which adds a portable diesel-driven battery charger to the DC system) identified in the FENOC letter dated June 24, 2011 (ML11180A233). LRA Appendix E, "Applicant's Environmental Report Operating License Renewal Stage," Attachment E, "Severe Accident Mitigation Alternatives Analysis," is revised to incorporate the corrected information in the affected Sections and Tables.

See the Enclosure to this letter for the revision to the Davis-Besse LRA.

Review for Impacts to Responses to Requests for Additional Information for the Review of the Davis Besse Nuclear Power Station, Unit No. 1 (Davis-Besse), License Renewal Application, Environmental Report, Attachment E, Severe Accident Mitigation Alternatives Analysis
Submitted by FENOC Letter Dated June 24, 2011 (ML11180A233)

FirstEnergy Nuclear Operating Company (FENOC) performed a review, based on the revised SAMA Analysis, for impacts to the responses to Nuclear Regulatory Commission (NRC) requests for additional information (RAIs) for the Severe Accident Mitigation Alternatives (SAMA) Analysis submitted by FENOC letter dated June 24, 2011 (ML11180A233). Based on the changes to the SAMA Analysis, no revision to the FENOC responses provided by the June 24, 2011, letter is necessary for the following SAMA RAIs:

SAMA RAI Responses – No Revision		
1.a	3.b	6.b
1.b	4.a	6.c
1.c	4.b	6.d
1.d	4.c	6.e
1.e	5.a	6.f
1.f	5.b	6.g
2.a	5.d.i	6.h
2.b	5.e	6.i.i
2.c	5.f	7.b
2.d	5.h	7.c
2.e	5.i	
3.a	6.a	

FENOC responses are revised for the remaining SAMA RAIs from the June 24, 2011, letter as provided in the following discussion. The NRC request is shown in bold text followed by the original FENOC response provided by the June 24, 2011, letter. A statement is provided for each SAMA RAI response to identify whether the response is replaced in its entirety or edited. For edited responses, the sentence affected is printed in *italics* with deleted text ~~*lined-out*~~ and added text *underlined*.

Item 3

Provide the following information with regard to the treatment and inclusion of external events in the SAMA analysis:

Question RAI 3.c

ER Section E.3.1.2.4 presents the basis for an external events multiplier of 3 based on a "conservatively" estimated fire CDF of 2.5E-05/yr developed using the FIVE methodology and the assumption that a "realistic" fire CDF is a factor of 3 less than this FIVE-produced fire CDF. The NRC staff disagrees that a fire CDF produced using the FIVE screening methodology is necessarily conservative in light of more recent research and guidance on hot short probabilities (i.e., NUREG/CR-6850). The NRC staff particularly notes that the minimal or non-treatment of hot shorts in the IPEEE FIVE analysis may more than offset other conservatisms in the FIVE analysis. Based on this, and the previous RAI, the NRC staff believes the best estimate of the fire CDF for Davis-Besse is 2.9E-05/yr. In addition, the USGS issued updated seismic hazard curves for much of the U.S. in 2008. Using this data, the NRC staff estimated a "weakest link model" seismic CDF for Davis-Besse of 6.7E-06/yr (see NRC Information Notice 2010-18 regarding Generic Issue 199). Based on a fire CDF of 2.9E-05/yr, a seismic CDF of 6.7E-06/yr, and an internal events CDF of 9.8E-06/yr, the NRC staff estimates the external events multiplier to be 3.6. In light of this, provide a revised SAMA evaluation using an external events multiplier of 3.6 or alternatively provide justification for an evaluation of a different multiplier based on this updated USGS information.

RESPONSE RAI 3.c

[The response to RAI 3.c is edited as shown in 2nd paragraph.]

Based on the information provided in the RAI, an updated external events multiplier was calculated for Davis-Besse. The updated external events multiplier includes risk contribution from fire, seismic, and other hazard groups. The risk contribution for the fire and seismic hazard groups was determined by a ratio between the hazard group CDF and the internal events CDF as shown in the equations below. The risk contribution from the other hazard group was conservatively assumed to be equivalent to the internal events contribution. Therefore, the other hazard group multiplier is 1.0.

Fire Hazard Multiplier:

$$\frac{\text{Fire CDF}}{\text{Internal Events CDF}} = \frac{2.9\text{x}10^{-5}/\text{yr}}{1.0\text{x}10^{-5}/\text{yr}} = 2.90$$

Seismic Hazard Multiplier:

$$\frac{\text{Seismic CDF}}{\text{Internal Events CDF}} = \frac{6.7\text{x}10^{-6}/\text{yr}}{1.0\text{x}10^{-5}/\text{yr}} = 0.67$$

To determine the multiplier to account for fire, seismic, and other hazard groups, the three individual multipliers were summed, resulting in a multiplier of 4.6. The cost-benefit evaluation was updated using an external event multiplier of 4.6. *The updated maximum benefit for Davis-Besse is ~~$1,955,223~~ $2,053,481.* Based on the updated maximum benefit, one SAMA candidate, AC/DC-03 (add a portable diesel-driven battery charger to the direct current (DC) system) was determined to be cost-beneficial.

ER Section E.3.1.2.4, “External Event Severe Accident Risk,” is deleted based on the response to this RAI. ER Section E.4.5, “Total Cost of Severe Accident Risk,” is revised to explain the updated external events multiplier. ER Tables E.4-1, E.7-2, E.7-3, E.7-5, and E.8-1 are revised to reflect the revised cost-benefit results.

See the Enclosure to this letter for the revision to the Davis-Besse LRA.

Item 4

Provide the following information concerning the Level 3 analysis:

Question RAI 4.d

ER Section E.3.4.6.2 does not identify the population base/year reference for the emergency planning zone (EPZ) evacuation speed. Describe how/whether the EPZ evacuation time was corrected for the year 2040 population (and address the population discrepancy noted in RAI 4.b).

RESPONSE RAI 4.d

[The response to RAI 4.d is edited as shown in the 2nd paragraph and Table 4.d-1.]

Reference [4] (in Attachment E of the Environmental Report) does not identify a collection date for the data that were used to estimate the evacuation speed in Section E.3.4.6.2. The evacuation information provided in Reference [4] was assumed to be current as of the 2000 census. However, no correction factor was applied to account for the increased population in 2040 in the original analysis.

Assuming that an increase in population is proportional to a decrease in evacuation speed, the evacuation speed was adjusted from 0.58 meters/second to 0.52 meters/second. This adjustment represents a 9.6 percent decrease in the evacuation speed, which was used to offset a 9.6 percent [$(1.047)^2 = 1.096$] increase in population at the end of the two-decade license renewal period. This decrease in evacuation speed was evaluated as a new sensitivity case (Sensitivity Case E3). *The results are provided in Table 4.d-1, below, and show no ~~very little~~ change from the base case, indicating that the results are not sensitive to slow evacuation speeds.* The base case results shown in Table 4.d-1 include the updated population (as needed to respond to RAI 4.b); similarly, sensitivity case E3 includes the updated population, to permit an equitable comparison to the base case. *ER Section E.3.5.2.4 is revised and new ER Table E.3-33, "Comparison of Base Case and Case E3," is added to incorporate sensitivity case E3.*

See the Enclosure to this letter for the revision to the Davis-Besse LRA.

Table 4.d-1: Comparison of Base Case and Case E3

	Internal Events		
	Base	**E3**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	~~*2.30E+00*~~ *2.12E+00*	~~*2.31E+00*~~ *2.12E+00*	~~*0.4%*~~ *0.0%*
Economic Impact (50) ($/yr)	~~*1.80E+03*~~ *3.59E+03*	~~*1.80E+03*~~ *3.59E+03*	0.0%

Question RAI 4.e

In ER Section E.3.5.2.3, for Case A1, identify the heat release energy (e.g. thermal, 1 MW) assumed for both the base and sensitivity cases.

RESPONSE RAI 4.e

[Table 4.e-1 is replaced in its entirety.]

The energy of release for the base case and sensitivity Case A1 are provided for each release category in Table 4.e-1, below.

Table 4.e-1 Energy of Release: Base Case and Sensitivity Case A1

Release Category	PLHEAT/Energy of Release (watts)	
	Base Case	Sensitivity Case A1
1.1	3.87E+07	1.49E+08
1.2	1.45E+07	9.21E+07
1.3	3.87E+07	1.49E+08
1.4	1.45E+07	9.21E+07
2.1	8.91E+06	6.04E+08
2.2	6.68E+06	6.16E+08
3.1	2.22E+06	2.92E+07
3.2	2.61E+06	1.82E+07
3.3	2.22E+06	1.78E+07
3.4	2.61E+06	1.82E+07
4.1	9.17E+05	1.66E+07
4.2	2.24E+05	1.66E+07
4.3	6.77E+05	1.66E+07
4.4	2.10E+05	1.66E+07
5.1	3.17E+06	2.48E+07
5.2	1.09E+07	6.31E+07
5.3	2.83E+06	2.01E+07
5.4	9.59E+06	5.80E+07
6.1	7.35E+07	3.36E+08
6.2	1.14E+08	4.64E+08
6.3	6.10E+07	3.87E+08
6.4	1.16E+08	4.90E+08
7.1	3.02E+07	1.79E+08
7.2	2.79E+07	1.67E+08
7.3	2.82E+07	1.68E+08
7.4	2.80E+07	1.66E+08
7.5	2.01E+07	1.34E+08
7.6	2.36E+07	1.22E+08
7.7	1.93E+07	1.32E+08
7.8	2.45E+07	1.29E+08
8.1	8.71E+06	1.61E+08
8.2	9.78E+07	4.11E+08
9.1	2.63E+02	2.08E+03
9.2	3.30E+02	2.14E+03

Item 5

Provide the following with regard to the SAMA identification and screening process:

Question RAI 5.c

None of the SAMA candidates identified in Table E.5-4 appear to be plant-specific SAMAs identified from plant-specific risk insights based on the current PRA model. Clarify how the importance lists were used to develop plant-specific SAMA candidates and justify the apparent absence of any plant-specific SAMA candidates. Also, the basic events identified in importance analysis Tables E.5-2 and E.5-3 are not linked to SAMA candidates. Sections E.5.4 and E.5.5 only discuss the SAMA candidates identified to address basic events with high risk reduction worth (RRW) values. Identify, for each basic event having a RRW benefit value (averted cost risk) greater than the minimum cost of a procedure change at Davis-Besse, the specific SAMA(s) that address each event and describe how the SAMA(s) address the basic event. Identify and evaluate SAMAs for basic events not addressed by an existing SAMA (e.g., flooding related basic events and initiators, including WHAF3ISE, SHAF2ISE, F3AM, and F7L). For any basic event for which no SAMA is identified, provide justification for not identifying a SAMA(s).

RESPONSE RAI 5.c

[The response to RAI 5.c is edited as shown on pages 9 and 10 of 63, and Table 5.c-2 on page 22 of 63. Tables 5.c-1 and 5.c-2 are also revised (multiple locations) to change the steam generator replacement schedule from 2013 to 2014 to align with current FENOC plans and with the discussions in the ER.]

The final list of SAMA candidates was developed from a combination of generic data, industry SAMA analyses and Davis-Besse-specific insights. The following SAMA candidates were added to the generic list based on Davis-Besse PRA-identified insights:

- SAMA candidate AC/DC-25 (dedicated DC power for AFW) and AC/DC-26 (alternator/generator for turbine-driven auxiliary feedwater (TDAFW) pump) were designed to extend the life of the TDAFW pumps in a station blackout (SBO) event and improve the likelihood of successful restoration of alternating current (AC) power.

- SAMA candidate AC/DC-27 (increased size of SBO fuel oil tank) was also designed to help mitigate an SBO event.

- SAMA candidate CB-21 (pressure sensors between the two in-series Decay Heat Removal (DHR) System suction valves) was designed to help reduce the likelihood of ISLOCA events.

- SAMA candidate CC-19 (automatic switchover of high pressure injection (HPI) and low pressure injection (LPI) suction from the BWST to the containment sump) was designed to increase the reliability of the switchover during a loss of coolant accident (LOCA) event.

- SAMA candidate CC-20 (modify hardware and procedures to allow using make-up pumps for high pressure recirculation from the containment sump) was designed improve the reliability of high pressure recirculation following the loss of HPI.

- SAMA candidate CC-21 (reduce the BSWT level at which switchover to containment recirculation is initiated) was designed to extend the time available to accomplish BWST refill.

- SAMA candidate CP-19 (install a redundant containment fan system) was designed to increase containment heat removal ability. This SAMA candidate was added as a variation to CP-18 to provide a redundant containment cooling function, in the form of containment fan coolers.

- SAMA candidates CW-24 (adding a diversified CCW pump) and CW-25 (providing the capability to cool makeup pumps with fire water on loss of CCW) were designed to mitigate the total loss of CCW cooling.

- SAMA candidate FW-16 (surveillance of manual AFW suction valves) was designed to improve the reliability of alternate sources of AFW water supply.

- SAMA candidate HV-06 (procedure guidance for alternate means of switchgear cooling) was designed to prevent the loss of one train of service water in the event of loss of one HVAC fan for the service water pump room. This SAMA candidate was developed from Davis-Besse IPE insights.

Evaluating Basic Events with Potential Benefit Greater Than the Cost of a Procedure Change

The internal events and LERF basic events with an RRW value estimated to be equal to or greater than the cost of a procedure change were evaluated. These basic events were dispositioned by either identifying resulting SAMAs or presenting the reason for no new SAMA candidate. One new SAMA candidate (OT-9R) resulted from this evaluation.

An estimate of the cost-benefit versus RRW was developed for the internal events basic events calculated for the base PRA model. The minimum cost of a procedure change was assumed to be $10,000. In addition, the minimum cost of a hardware modification was estimated to be $100,000. The cost-benefit versus RRW assumed that cost-benefit was directly proportional to the reduction in core damage frequency (CDF). Cost is not perfectly correlated with CDF, due to the fact that different scenarios, even with the same CDF, will result in different distributions of release categories. It is judged, however, that this correlation provides a reasonable estimate of potential benefit along with what is judged to be a low cost for a procedure change, and provides strong confidence that cost-effective SAMA candidates will be captured.

For the total benefit for the hazard group (Bt), the cost-benefit versus RRW used the maximum derived benefit of ~~$349,147~~ $366,693.

The following formula is used for deriving the estimated benefit by hazard group based on RRW:

$$EB_{(BE)} = B_t\left(1 - \frac{1}{RRW}\right)$$

where,

$EB_{(BE)}$ = the estimated benefit based on a basic event

B_t = the total benefit for the hazard group (internal events, fire, or seismic)

RRW = the RRW for the basic event from the PSA, by hazard, assuming the basic event failure probability is reduced to zero.

The RRW for the Level 2 PRA basic events may be calculated based on LERF rather than CDF. Additional conservatism is added by treating Level 2 PRA basic event RRW values based on LERF as if they were based on CDF (i.e., the use of B_t significantly overstates their benefit), and the degree of conservatism could be large.

Based on these estimates, an RRW value of 1.03 was calculated to have a maximum cost benefit of $10,000 and an RRW of ~~1.40~~ 1.37 was estimated to have a maximum cost benefit of $100,000. The maximum cost benefit is based on the RRW of the basic event being reduced to 1.0 (basic event modeled as perfect). For all basic events having an RRW value estimated to be at, or above, the value of a procedure change, a disposition was provided either identifying the SAMA candidate(s) addressing that basic event or a description as to why the basic event was not addressed in a SAMA candidate. No basic events had an RRW value equal to, or greater than the estimated cost of a hardware modification. Table 5.c-1, below, lists the basic events with the highest RRW for CDF.

Table 5.c-2, below, tabulates the basic events with the highest RRW for LERF. The estimated benefit for each basic event was derived by taking the RRW for LERF and applying the maximum total benefit used for the CDF basic events. This is very conservative, since the total maximum benefit does not apply only to LERF. For all basic events having an RRW value estimated to be at, or above, the value of a procedure change, a disposition was provided either identifying the SAMA candidate(s) addressing that basic event or a description as to why the basic event was not addressed in a SAMA candidate. No basic events had an RRW value equal to, or greater than the estimated cost of a hardware modification.

Basic events WHAF3ISE, SHAF2ISE, F3AM, and F7L did not have RRW values with potential benefit equal to, or greater than, the minimum cost of a procedure change. Basic event F7L, a large circulating water flood in the Turbine Building, did, however, result in an RRW value greater than the minimum cost of a procedure change for the 95 percent uncertainty CDF model. SAMA candidate FL-01 (improve inspection of rubber expansion joints on main condenser) was initially identified to address basic event F7L, and was designed to reduce the frequency of a large circulating water system flooding event due to failure of the circulating water system expansion joints. Based on the F7L RRW value from the 95 percent uncertainty CDF model and its original screening of “Very Low Benefit,” SAMA candidate FL-01 was reevaluated and screened as “Already Implemented,” as discussed in the response to RAI 6.k.

The ER is revised (numerous locations) to identify that there are now 168 SAMA candidates that were evaluated instead of the original 167. Also, ER Table E.5-4 is revised to include changes identified in Tables 5.c-1 and 5.c-2, below.

See the Enclosure to this letter for the revision to the Davis-Besse LRA.

Table 5.c-1 – Basic Event Level 1 PRA Importance				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
UHAMUHPE	2.59E-01	1.349	Failure to initiate makeup/HPI cooling after loss of all feedwater	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.
QHAMDFPE	2.45E-01	1.324	Failure to start MDFP after loss of feedwater	SAMA candidate FW-17R evaluates implementing an automatic start of the motor-driven feed pump (MDFP) on loss of main feedwater (MFW).
QHARCPCE	2.32E-01	1.302	Operators fail to trip RCPs after a total loss of seal cooling	SAMA candidate CW-26R evaluates implementing an automatic RCP trip on high bearing cooling temperature or loss of CCW flow to the RCP thermal barrier cooler and loss of seal injection flow.
T3	1.96E-01	1.243	LOOP (initiating event)	Numerous SAMA candidates that address LOOP were evaluated: AC/DC-01, additional battery capacity AC/DC-14, install gas turbine generator AC/DC-25, provide dedicated DC system to TDAFW pumps AC/DC-26, provide alternator/generator driven by TDAFW pumps AC/DC-27, increase SBO fuel oil tanks size
EHASBDGE	1.64E-01	1.196	Operators fail to align power from SBO diesel generator to supply MDFP	SAMA candidate AC/DC-28R evaluates the automatic start of the SBO diesel and loading to Bus D2 upon loss of power to Bus D2.

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
EHASBD1E	1.58E-01	1.187	Operators fail to start SBO diesel generator and align to bus D1	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.
EHAD2DGE	1.53E-01	1.181	Operators fail to align power from EDG 1-1 or EDG 1-2 to supply MDFP given LOOP	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
T1	1.35E-01	1.156	Reactor/turbine trip (initiating event)	This is based on a somewhat conservative T1 value of 1.02/yr. Davis-Besse trip occurrence frequency is considered representative of industry values. SAMA candidates have been evaluated that address various Davis-Besse important scenarios following a reactor/turbine trip. CC-01, evaluates the installation of an independent active or passive HPI system. CW-26R, evaluates an automatic RCP trip on high motor bearing temperature or loss of CCW flow to the RCP thermal barrier cooler and loss of seal injection flow. FW-17R, evaluates an automatic start of the motor driven feedwater pump. HV-01, evaluates a redundant train for ventilation. HV-03, evaluates the staging of backup fans in the switchgear room.
QHAOVF2E	1.22E-01	1.139	Operators fail to take local manual control of TDAFW pump 1-2 speed.	SAMA candidate AC/DC-25 provides a dedicated DC system to TDAFW pumps and SAMA candidate AC/DC-26 provides an alternator/generator driven by TDAFW pumps. These SAMA candidates would eliminate the need for local manual control of the TDAFW pumps.

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
ZHARCPCE	1.10E-01	1.124	Operators fail to trip RCPs following loss of seal cooling	SAMA candidate CW-26R evaluates implementing an automatic RCP trip on high bearing cooling temperature or loss of CCW flow to the RCP thermal barrier cooler and loss of seal injection flow.
WHASPREE	1.07E-01	1.12	Failure to recover CCW using spare CCW train (prior to damage)	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.
QMBAFP11	7.61E-02	1.082	AFW Train 1 in maintenance	This estimated benefit of this basic event is below the minimum estimated cost of a hardware modification. The following SAMA candidates address improvements to the reliability of AFW in loss of off-site power scenarios: AC/DC-25, provide dedicated DC system to TDAFW pumps AC/DC-26, provide alternator/generator driven by TDAFW pumps
XHOS-CCW1RUN2STBY	7.54E-02	1.082	CCW Pump 1 running, Pump 2 in standby	This is a plant configuration probability in the model. It does not contribute to risk.
EDG0012F	7.12E-02	1.077	EDG 1-2 fails to run	SAMA candidate AC/DC-14 evaluates adding a gas turbine generator as an additional source of on-site power.

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
ZOP007BR	7.09E-02	1.076	Failure to restore off-site power	Numerous SAMA candidates that address LOOP were evaluated: AC/DC-01, additional battery capacity AC/DC-14, install gas turbine generator AC/DC-25, provide dedicated DC system to TDAFW pumps AC/DC-26, provide alternator/generator driven by TDAFW pumps AC/DC-27, increase SBO fuel oil tanks size
TMPP43XF-CC_ALL	6.79E-02	1.073	All CCW pumps fail to run due to CCF (initiating event)	SAMA candidate CW-24 evaluates the standby CCW pump with a pump diverse from the other two CCW pumps.
XHOS-CCW2RUN1STBY	6.57E-02	1.07	CCW Pump 2 running, Pump 1 in standby	This is a plant configuration probability in the model. It does not contribute to risk.
R	6.37E-02	1.068	SGTR (initiating event)	*Davis-Besse is scheduled to install new steam generators in ~~2013~~ 2014*. This modification, with resulting reduction in SGTR frequency, is not reflected in the current PRA model. This plant improvement is assumed to result in a reduction risk importance of SGTR events.
EHAD1ACE	5.90E-02	1.063	Failure to lineup alternate source to D1	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
T2	5.86E-02	1.062	Plant trip due to loss of MFW (initiating event)	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
NORCVRT3	5.57E-02	1.059	Offsite power recovery not possible after a tornado.	Numerous SAMA candidates that address LOOP were evaluated: AC/DC-01, additional battery capacity AC/DC-14, install gas turbine generator AC/DC-25, provide dedicated DC system to TDAFW pumps AC/DC-26, provide alternator/generator driven by TDAFW pumps AC/DC-27, increase SBO fuel oil tanks size
AV	5.12E-02	1.054	Reactor vessel rupture	Reactor vessel rupture is a low probability event that that is assumed to result in guaranteed core damage. No applicable SAMA candidates were considered possible to prevent core damage.
QTP000XA-CC_1_2	5.13E-02	1.054	CCF of two components: QTP0001A & QTP0002A (TDAFW)	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
QTP0001A	4.90E-02	1.051	AFP/T-1 fails to start	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
QMBAFP12	4.67E-02	1.049	AFW Train 2 in maintenance	The estimated benefit for this basic event is below the cost of a hardware modification. The following SAMA candidates address improvements to the reliability of AFW in LOOP scenarios: AC/DC-25, provide dedicated DC system to TDAFW pumps AC/DC-26, provide alternator/generator driven by TDAFW pumps
ZOP006FR	4.58E-02	1.048	Failure to restore off-site power	Numerous SAMA candidates that address LOOP were evaluated: AC/DC-01, additional battery capacity AC/DC-14, install gas turbine generator AC/DC-25, provide dedicated DC system to TDAFW pumps AC/DC-26, provide alternator/generator driven by TDAFW pumps AC/DC-27, increase SBO fuel oil tanks size
S	4.35E-02	1.045	Small LOCA (initiating event)	SAMA candidate CC-01 evaluates the installation of an independent active or passive HPI system. SAMA candidate CC-19 evaluates the implementation of automatic switchover of HPI and LPI suction from the BWST to the to containment sump for LOCAs.

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
T13A-1-3-IEF	4.18E-02	1.044	Loss of CCW Train 1 initiating event Pump 1 running	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
MHARMVTE	4.17E-02	1.043	Operators fail to compensate for loss of room cooling for makeup pumps.	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.
XHAMUCDE	4.10E-02	1.043	Operators fail to attempt cooldown via makeup/HPI cooling.	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.
T13A-2-3-IEF	3.93E-02	1.041	Loss of CCW Train 2 initiating event Pump 2 running	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
EMBEDG12	3.85E-02	1.04	EDG Train 2 in maintenance	SAMA candidate AC/DC-14 evaluates adding a gas turbine generator as an additional source of on-site power.

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
CHASGDPE	3.63E-02	1.038	Operators fail to cooldown during a SGTR	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R. *Also, Davis-Besse is scheduled to install new steam generators in ~~2013~~ 2014*. This modification, with resulting reduction in SGTR frequency, is not reflected in the current PRA model. This plant improvement is assumed to result in a reduction risk importance of SGTR events.
FMFWTRIP	3.71E-02	1.038	MFW/ICS faults following trip	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
FMM00003	3.52E-02	1.037	Any MSSVs on SG1 fail to reseat	SAMA candidate CB-22R evaluates the use of a “gagging device” to close a stuck open MSSV.
EDG0012A	3.46E-02	1.036	EDG 1-2 fails to start	SAMA candidate AC/DC-14 evaluates adding a gas turbine generator as an additional source of on-site power.
AASGTR11	3.42E-02	1.035	SGTR occurs on OTSG 1-1 (split fraction)	*Davis-Besse is scheduled to install new steam generators in ~~2013~~ 2014*. This modification, with resulting reduction in SGTR frequency, is not reflected in the current PRA model. This plant improvement is assumed to result in a reduction risk importance of SGTR events.

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
LHAMSIVE	3.34E-02	1.035	Failure to close MSIV and isolate steam generator containing ruptured tube	*Davis-Besse is scheduled to install new steam generators in ~~2013~~ 2014*. This modification, with resulting reduction in SGTR frequency, is not reflected in the current PRA model. This plant improvement is assumed to result in a reduction risk importance of SGTR events.
QHAMDF3E	3.34E-02	1.035	Failure to start MDFP prior to depletion of BWST during makeup	SAMA candidate FW-17R evaluates implementing an automatic start of the motor-driven feed pump (MDFP) on loss of main feedwater (MFW). SAMA candidate CC-22R evaluates implementing an automatic refilling of the BWST.
QTP0002A	3.25E-02	1.034	AFP/T-2 fails to start	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
EDG0011F	3.13E-02	1.032	EDG 1-1 fails to run	SAMA candidate AC/DC-14 evaluates adding a gas turbine generator as an additional source of on-site power.
FCIRCTMP	3.00E-02	1.031	Circ water temperature not acceptable	This is a PRA model flag. It is not a candidate for a SAMA. No SAMA candidate considered.
RRW of 1.03 is estimated to have a cost of approximately $10,000. This is assumed to be the minimum cost of a procedure change.				

Table 5.c-2 – Basic Event LERF Importance				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
R	9.00E-01	10.048	SGTR (initiating event)	*Davis-Besse is scheduled to install new steam generators in ~~2013~~ 2014*. This modification, with resulting reduction in SGTR frequency, is not reflected in the current PRA model. This plant improvement is assumed to result in a reduction risk importance of SGTR events.
XHAMUCDE	6.10E-01	2.563	Operators fail to attempt cooldown via makeup/HPI cooling	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.
CHASGDPE	5.40E-01	2.175	Operators fail to cooldown during a SGTR	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.
LHAMSIVE	4.97E-01	1.989	Failure to close MSIV and isolate steam generator containing ruptured tube	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.

Table 5.c-2 – Basic Event LERF Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
AASGTR11	4.81E-01	1.926	SGTR occurs on OTSG 1-1 (split fraction)	*Davis-Besse is scheduled to install new steam generators in ~~2013~~ 2014*. This modification, with resulting reduction in SGTR frequency, is not reflected in the current PRA model. This plant improvement is assumed to result in a reduction risk importance of SGTR events.
AASGTR12	3.93E-01	1.646	SGTR occurs on OTSG 1-2 (split fraction)	*Davis-Besse is scheduled to install new steam generators in ~~2013~~ 2014*. This modification, with resulting reduction in SGTR frequency, is not reflected in the current PRA model. This plant improvement is assumed to result in a reduction risk importance of SGTR events.
RRW of ~~1.40~~ 1.37 is estimated to have a cost of approximately $100,000. **This is assumed to be the minimum cost of a hardware modification.**				
FMM00003	7.90E-02	1.086	Any MSSVs on SG1 fail to reseat	SAMA candidate CB-22R evaluates the use of a "gagging device" to close a stuck open MSSV.
VD-IEF	7.54E-02	1.082	ISLOCA due to internal rupture of DHR suction valves	SAMA candidate CB-21 evaluates placing pressure measurements between the two DHR suction valves in the RCS hot leg allowing early detection of inboard isolation valve leakage.
FLCO101F	7.31E-02	1.079	Logic card fails during operation – MSIV 101 fails to close	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.

Table 5.c-2 – Basic Event LERF Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
LPPNISOZ	7.18E-02	1.077	ISLOCA occurs in non-isolable portion of DHR system	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
FMM00004	6.80E-02	1.073	Any MSSVs on SG2 fail to reseat	SAMA candidate CB-22R evaluates the use of a “gagging device” to close a stuck open MSSV.
FLC0100F	6.13E-02	1.065	Logic card fails during operation – MSIV 100 fails to close	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
QHAMDFPE	5.96E-02	1.063	Failure to start MDFP as backup to turbine-driven feedwater pumps for transient, Small LOCA or SGTR events	SAMA candidate FW-17R evaluates implementing an automatic start of the motor-driven feed pump (MDFP) on loss of main feedwater (MFW).
EC1ZXXXN-CC_1_2	5.19E-02	1.055	CCF of two components: EC1Z089N & EC1Z100N	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
LPSRC2BH	4.93E-02	1.052	Press switch PSH RC2B4 fails high – fails DHR	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
LPSZ416H	4.93E-02	1.052	Press switch PSH 7531A fails high - fails DHR	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.

Table 5.c-2 – Basic Event LERF Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
LMVF012R	4.53E-02	1.047	Internal rupture of DH 12 (annual frequency)	SAMA candidate CB-21 evaluates placing pressure measurements between the two DHR suction valves in the RCS hot leg allowing early detection of inboard isolation valve leakage.
LMBCWRT1	4.12E-02	1.043	CWR Train 1 unavailable due to maintenance	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
EDG0012F	3.47E-02	1.036	EDG 1-2 fails to run	SAMA candidate AC/DC-14 evaluates adding a gas turbine generator as an additional source of on-site power.
FCIRCTMP	3.00E-02	1.031	Circ water temperature not acceptable	This is a PRA model flag. It is not a candidate for a SAMA. No SAMA candidate considered.
FVV011BT	3.04E-02	1.031	AVV ICS11B fails to reseat after steam	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
LMVF011R	3.01E-02	1.03	Internal rupture of DH 11 (annual frequency)	SAMA candidate CB-21 evaluates placing pressure measurements between the two DHR suction valves in the RCS hot leg allowing early detection of inboard isolation valve leakage.

Table 5.c-2 – Basic Event LERF Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
ELOOPRT	2.93E-02	1.03	LOOP given reactor trip	Numerous SAMA candidates that address LOOP were evaluated: AC/DC-01, additional battery capacity AC/DC-14, install gas turbine generator AC/DC-25, provide dedicated DC system to TDAFW pumps AC/DC-26, provide alternator/generator driven by TDAFW pumps AC/DC-27, increase SBO fuel oil tanks size
RRW of 1.03 is estimated to have a cost of approximately $10,000. This value is assumed to be the minimum cost of a procedure change.				

Question RAI 5.d

ER Section E.5.3, E.5.4, and E.5.5 discuss significant contributors to core damage frequency (CDF) and large early release frequency (LERF). These sections and the associated tables show that there are a number of operator errors and non-recovery actions that occur in these listings, but report that no weaknesses in training or procedures were identified. Given: 1) the significant number of operator errors in these lists, 2) that human errors are among the most dominant failure modes presented in the importance Tables E.5-2 (i.e., the first 9 basic events listed by RRW are human error events) and E.5-3, and 3) that operator errors often have relatively high failure probabilities, provide the following:

i. **Explain the process used to make the determination that there were no opportunities to improve procedures and training.**

ii. **Discuss whether any of the risk significant operator action failures could be addressed by a SAMA to automate the function (i.e., automating tripping of the RCPs after a loss of seal cooling -see RAI 7.a).**

RESPONSE RAI 5.d

[The response to RAI 5.d.ii is edited as shown in the middle of the 1st paragraph. Also, Tables 5.d-1, 5.d-2 and 5.d-3 are replaced in their entirety.]
[NOTE: * One sensitivity case no longer applies and has been deleted; see the revised response to RAI 6.i.ii]

5.d.ii

In addition to the new SAMAs addressed in RAI 7, two additional SAMA candidates were evaluated to address automating risk significant operation actions: SAMA candidate AC/DC-28R (automatically start and load the SBODG on Bus D2 upon loss of power to the bus), and SAMA candidate OT-08R (automatically start and load the SBODG on Bus D2 upon loss of power to the bus in combination with automatically starting the MDFP). Table 5.d-1 and Table 5.d-2, below, provide the internal event and total benefit results for SAMA candidates AC/DC-28R and OT-08R, respectively. *Table 5.d-3, below, provides the final results for the ~~ten~~ nine* [*] *sensitivity cases for SAMA candidate AC/DC-28R and OT-08R.* The implementation cost for SAMA candidate AC/DC-28R was estimated as $1,600,000. Therefore, this SAMA candidate is not cost-beneficial at Davis-Besse. The implementation cost for SAMA candidate OT-08R was estimated as $4,400,000. Therefore, this SAMA candidate is not cost-beneficial at Davis-Besse.

Table 5.d-1: Internal Events Benefit Results for SAMA Candidates AC/DC-28R and OT-08R

Case	**AC/DC-28R** (Auto SBODG)	**OT-08R** (Auto SBODG & MDFP)
Off-site Annual Dose (rem)	2.03E+00	1.92E+00
Off-site Annual Property Loss ($)	3.45E+03	3.26E+03
Comparison CDF	1.0E-05	1.0E-05
Comparison Dose (rem)	2.12E+00	2.12E+00
Comparison Cost ($)	3.59E+03	3.59E+03
Enhanced CDF	8.3E-06	5.7E-06
Reduction in CDF	**17.00%**	**43.00%**
Reduction in Off-site Dose	**4.25%**	**9.43%**
Immediate Dose Savings (On-site)	$138	$348
Long Term Dose Savings (On-site)	$600	$1,518
Total Accident Related Occupational Exposure **(AOE)**	$738	$1,866
Cleanup/Decontamination Savings (On-site)	$22,502	$56,916
Replacement Power Savings (On-site)	$22,766	$57,584
Averted Costs of On-site Property Damage **(AOSC)**	$45,267	$114,500
Total On-site Benefit	**$46,005**	**$116,366**
Averted Public Exposure **(APE)**	$2,209	$4,908
Averted Off-site Damage Savings **(AOC)**	$1,718	$4,049
Total Off-site Benefit	**$3,926**	**$8,957**
Total Benefit (On-site + Off-site)	**$49,932**	**$125,323**

Table 5.d-2: Total Benefit Result for SAMA Candidates AC/DC-28R and OT-08R

	AC/DC-28R (Auto_SBODG)	**OT-08R** (Auto_SBODG & MDFP)
Internal Events	$49,932	$125,323
Fires, Seismic, Other	$229,685	$576,486
Total Benefit	$279,617	$701,809

Table 5.d-3: Final Results of the Sensitivity Cases for SAMA Candidates AC/DC-28R and OT-08R

SAMA ID	Repair Case	Low Discount Rate Case	High Discount Rate Case	On-site Dose Case	On-site Clean-up Case
AC/DC-28R	$177,626	$422,629	$193,345	$283,796	$321,619
OT-08R	$443,832	$1,060,578	$484,871	$712,381	$808,052

SAMA ID	Replacement Power Case	Multiplier Case	Evacuation Speed	95th CDF Case
AC/DC-28R	$365,190	$399,452	$279,617	$405,444
OT-08R	$918,258	$1,002,584	$701,809	$1,017,623

Question RAI 5.g

Several SAMA candidates identified in Table E.6-1 are subsumed in another SAMA candidate (e.g., AC/DC-06, AC/DC-09, AC/DC-20). For each subsumed SAMA candidate, provide an assessment of its implementation cost relative to that of the SAMA into which it was subsumed. If the implementation cost of the subsumed SAMA is less, provide a revised basis for the Phase I screening and Phase II cost-benefit evaluation if it meets Criterion F.

RESPONSE RAI 5.g

[The response to RAI 5.g is edited as shown in Table 5.g-1.]

SAMA candidate CB-08 was subsumed in SAMA candidate CB-07 in Table E.6-1. SAMA candidate CB-07 was screened as 'already implemented' at Davis-Besse. The nature of the operation action/training is similar in both SAMA candidates. Therefore, SAMA candidate CB-08 was re-screened as Criterion B (Already Implemented). Accordingly, there was no need to determine the cost of implementation and assess the cost-benefit of SAMA candidate CB-08. ER Table E.6-1 is revised to identify the re-screening of SAMA candidate CB-08.

The SAMA candidates subsumed in Phase I (AC/DC-06, AC/DC-09, AC/DC-20, and CC-08) have an equivalent or higher cost of implementation than the SAMA candidates evaluated in Phase II. Nonetheless, an analysis was performed to assess the cost-benefit of the subsumed SAMA candidates. The total benefit was derived from the SAMA candidates into which they were subsumed and compared to the cost of implementation. Table 5.g-1 provides the results of the cost-benefit evaluation. None of the subsumed SAMA candidates are cost-beneficial to implement at Davis-Besse.

See the Enclosure to this letter for the revision to the Davis-Besse LRA.

Table 5.g-1: Final Results of the Cost-Benefit Evaluation for Subsumed SAMA Candidates

SAMA ID	Modification	Estimated Benefit	Cost Estimate	Conclusion
AC/DC-06	Provide additional DC power to the 120/240V vital AC system.	~~*$94,363*~~ *$100,547*	$1,750,000	Not Cost Effective
AC/DC-09	Provide an additional diesel generator.	~~*$94,363*~~ *$237,436*	$2,800,000	Not Cost Effective
AC/DC-20	Add a new backup source of diesel generator cooling.	~~*$33,745*~~ *$39,242*	$700,000	Not Cost Effective
CC-08	Add the ability to automatically align ECCS to recirculation mode upon BWST depletion.	$15,155	$1,500,000	Not Cost Effective

Item 6

Provide the following with regard to the Phase II cost-benefit evaluations:

Question RAI 6.i

ER Section E.8.6 discusses six sensitivity cases. Relative to these sensitivity cases, provide the following:

ii. The description of the sixth sensitivity case states that off-site economic cost was increased by 25 percent. Table E.8-1 indicates that the total benefit for each of the SAMA candidates was increased by the same amount of $19,632, the offsite economic cost (AOC) value. Clarify how the increase of 25 percent in off-site economic cost correlates to the increase in total benefits of $19,632 for each SAMA.

RESPONSE RAI 6.i

[The response to RAI 6.i.ii is replaced in its entirety.]

6.i.ii

The sensitivity case for which the off-site economic cost was increased by 25 percent has been removed as a sensitivity case as it is no longer germane, since the MACCS2 economic input values that formed the basis for the sensitivity case were increased to reflect 2009 dollars, the reference economic year for the SAMA analysis.

Question RAI 6.j

ER Section 8.3 discusses a sensitivity case using a higher evacuation speed. Provide the evacuation speed used for this analysis. Also, Table E.3-31 shows that the population dose decreased compared to the base case yet Table E.8-1 shows the total net benefit increased by $1,963 for each SAMA. Explain this anomalous result and describe the methodology for developing the $1,963 used for each SAMA.

RESPONSE RAI 6.j

[The response to RAI 6.j is edited as shown.]

The evacuation speed used in the sensitivity case discussed in ER Section E.8.3 was 1.0 meter/second. *The population dose risk used in the Section E.8.3 sensitivity case was the result of the Level 3 PRA sensitivity case E1.*

As noted in the RAI, with a decrease in population dose risk, the net benefit for each SAMA candidate would be expected to decrease. The anomalous result (e.g., a net benefit increase) was due to the number of significant figures used in the Level 3 PRA and the cost-benefit evaluation. *The population dose risk values differed in the third significant digit, which when rounded caused the unexpected results. As a result of the response to RAI 4.b, above, the population dose risk values have been revised for the Level 3 PRA sensitivity case E1. The ER revisions due to population dose risk were identified in the response to RAI 4.b.*

With the revised results from RAI 4.b and consistent use of significant figures between the Level 3 PRA and cost-benefit analysis, the value $1,963 is no longer germane to the sensitivity case in Section E.8.3.

As noted in the staff's RAI, a decrease in population dose risk was the result of sensitivity case E1 (where the evacuation speed was increased). Since NEI 05-01 suggested an evacuation speed sensitivity case to assess the impact on the results due to the uncertainty in the evacuation speed, it is logical to test (via a sensitivity case) the impact of a lower evacuation speed (which may cause a previously screened SAMA candidate to become cost-beneficial). *Accordingly, the cost-benefit sensitivity case (Evacuation Speed from Table E.8-1) has been revised to use the results from Level 3 PRA sensitivity case E3 (see response to RAI 4.d), in which the evacuation speed is decreased by 9.6 percent, which ~~causes a slight~~ results in no increase in population dose risk. ~~ER Section E.3.5.2.4 is revised and new ER Table E.3-33 is added to incorporate sensitivity case E3.~~*

The total benefit for each SAMA candidate ~~has been increased by $1374~~ did not increase, which is consistent with ~~the~~ no increase in population dose risk. For the sensitivity case in Section E.8.3, the population dose~~s~~ risk values are taken from the Level 3 PRA sensitivity case E3 and replace the base case values in the determination of the averted public exposure (APE). ~~Since there is a constant difference in the population dose values, for the Section E.8.3 sensitivity case, the total benefit for each SAMA is changed by the same dollar amount. (See Table E.8-1 for results of evacuation speed sensitivity case in response to RAI 4.b.)~~

~~See the Enclosure to this letter for the revision to the Davis-Besse LRA.~~

Question RAI 6.k

The ER provides no assessment of the uncertainty distribution for CDF. Relative to the uncertainty distribution, address the following:

- **Provide the uncertainty distribution (5th, mean, and 95th percentiles) for the Davis-Besse PRA model CDF and describe how the distribution was developed.**

- **Provide an assessment of whether an uncertainty analysis using the 95th percentile CDF and the external events multiplier of 3.6 developed in RAI 3.c is bounded by the Multiplier Case sensitivity analysis. If not bounded, provide an uncertainty analysis using the 95th percentile CDF. In this analysis, provide an assessment of each Phase 1 SAMA eliminated using Screening Criterion D and E to determine whether any Phase 1 SAMAs originally screened should have a Phase 2 cost-benefit evaluation performed. Provide a Phase 2 cost-benefit evaluation for any SAMA not screened.**

- **If the Multiplier Case is bounding, provide an assessment of each Phase 1 SAMA eliminated using Screening Criteria D and E to determine whether any Phase 1 SAMAs originally screened should have a Phase 2 cost-benefit evaluation performed. Provide a Phase 2 cost-benefit evaluation for any SAMA not screened.**

RESPONSE RAI 6.k

[The response to RAI 6.k is edited as shown in Tables 6.k-1 and 6.k-2. Table 6.k-3 is also revised (page 39 of 63) to change the steam generator replacement schedule from 2013 to 2014 to align with current FENOC plans and with the discussions in the ER.]

The following table provides the uncertainty distribution for the Davis-Besse SAMA PRA model CDF. The 5th, mean, and 95th percentile values are in **bold** font:

	5% Conf.	Mean	95% Conf.
Point Estimate		9.70E-06	
Mean	1.06E-05	**1.07E-05**	1.09E-05
5th percentile	7.18E-06	**7.20E-06**	7.22E-06
Median	9.51E-06	9.53E-06	9.55E-06
95th percentile	1.53E-05	**1.55E-05**	1.56E-05
StdDev		1.48E-05	

Skewness		5.75E+01	
Kurtosis		4.55E+03	

The SAMA analysis model database was modified to support performance of an uncertainty analysis using the UNCERT software package. Failure rate distributions were entered into the database and modifications were made to make the database compatible with the UNCERT software. The SAMA analysis level 1 model was re-quantified to provide a cutset file compatible with the UNCERT software, and the uncertainty analysis was performed using the revised cutset file and database.

An assessment of the impact of the 95th percentile CDF uncertainty for internal events was performed for Davis-Besse. The uncertainty factor was derived from a ratio of the 95th percentile CDF uncertainty (1.55E-05/yr) to the point estimate CDF (1.07E-05/yr) for internal events. The uncertainty factor used in this analysis was 1.45. The analysis also used an external events multiplier of 4.6 (see the response to RAI 3.c for additional information on the development of the external events multiplier). Table 6.k-1, below, provides the cost-benefit results for the 95th percentile CDF uncertainty factor case. Also, the Multiplier Case was updated using an external events multiplier of seven (7). Table 6.k-2, below, provides the Multiplier Case cost-benefit results. The results of the 95th percentile CDF uncertainty and Multiplier Case sensitivity analyses identified one SAMA candidate (AC/DC-03) to be cost effective.

Since the external event multiplier used in the base case and the sensitivity case have changed, the issue of bounding is no longer relevant. Nonetheless, the SAMA candidates designated as Criterion D (Very Low Benefit) were re-evaluated (see Table 6.k-3, below) based on the results of the 95th percentile CDF uncertainty. For SAMA candidates where the 95th percentile CDF uncertainty basic event data were available, these basic events' RRW data were used as a basis for the final determination. For some SAMA candidates, either basic event data were not available, or basic event data were not applicable to the determination; for those cases, the determination basis is also provided.

SAMA candidate FL-01 (improve inspection of rubber expansion joints on main condenser) was initially identified for cost-benefit analysis based on the 95th percentile CDF uncertainty results. However, upon further investigation, the disposition of SAMA candidate FL-01 is changed to Criterion B (Already Implemented). The basis for the revised disposition is that the circulating water joints are currently inspected during outages and periodically replaced. ER Table E.6-1 is revised to include this change.

Further, based on additional information, SAMA candidate OT-05 (increase training and operating experience feedback to improve operator response) is changed from Criterion D (Very Low Benefit) to Criterion B (Already Implemented). The basis for the revised disposition is that Davis-Besse provides PRA information, such as risk

significant initiating events, high worth operator actions and high worth equipment, to operators and other departments. Attachment 2 of FENOC procedure NOPM-CC-6000, "Probabilistic Risk Assessment Program," identifies items supported by the PRA Program; one item is PRA training support in areas such as new licensed operator training and operator re-qualification training cycles. ER Table E.6-1 is revised to include this change.

SAMA candidates screened with Criterion E (Subsumed) were addressed in the response to RAI 5.g, above.

See the Enclosure to this letter for the revision to the Davis-Besse LRA.

Table 6.k-1: 95th Percentile Uncertainty Factor Cost-Benefit Results

SAMA ID	95th Percentile Uncertainty Factor Estimated Benefit	Estimated Cost	Conclusion
AC/DC-01	*$145,794*	$1,750,000	Not Cost Effective
AC/DC-03	*$575,095*	$330,000	Cost Effective
AC/DC-14	*$344,283*	$2,000,000	Not Cost Effective
AC/DC-19	*$56,901*	$700,000	Not Cost Effective
AC/DC-21	*$68,912*	$100,000	Not Cost Effective
AC/DC-25	*$354,521*	$2,000,000	Not Cost Effective
AC/DC-26	*$354,521*	$2,000,000	Not Cost Effective
AC/DC-27	$0	$550,000	Not Cost Effective
CB-21	*$42,842*	$550,000	Not Cost Effective
CC-01	*$4,982*	$6,500,000	Not Cost Effective
CC-04	$0	$5,500,000	Not Cost Effective
CC-05	$0	$6,500,000	Not Cost Effective
CC-19	$21,974	$1,500,000	Not Cost Effective
HV-01	*$1,993*	$50,000	Not Cost Effective
HV-03	*$1,993*	$400,000	Not Cost Effective
AC/DC-28R	*$405,444*	$1,600,000	Not Cost Effective
CB-22R	*$162,566*	$4,600,000	Not Cost Effective
CC-22R	$0	$2,200,000	Not Cost Effective
CW-26R	*$529,319*	$1,500,000	Not Cost Effective
FW-17R	*$592,197*	$2,800,000	Not Cost Effective
OT-08R	*$1,017,623*	$4,400,000	Not Cost Effective

Table 6.k-2: Multiplier Case Cost-Benefit Results

SAMA ID	Multiplier Case	Estimated Cost	Conclusion
AC/DC-01	*$143,639*	$1,750,000	Not Cost Effective
AC/DC-03	*$566,596*	$330,000	Cost Effective
AC/DC-14	*$339,195*	$2,000,000	Not Cost Effective
AC/DC-19	*$56,060*	$700,000	Not Cost Effective
AC/DC-21	$67,893	$100,000	Not Cost Effective
AC/DC-25	*$349,282*	$2,000,000	Not Cost Effective
AC/DC-26	*$349,282*	$2,000,000	Not Cost Effective
AC/DC-27	$0	$550,000	Not Cost Effective
CB-21	*$42,209*	$550,000	Not Cost Effective
CC-01	*$4,908*	$6,500,000	Not Cost Effective
CC-04	$0	$5,500,000	Not Cost Effective
CC-05	$0	$6,500,000	Not Cost Effective
CC-19	$21,649	$1,500,000	Not Cost Effective
HV-01	*$1,963*	$50,000	Not Cost Effective
HV-03	*$1,963*	$400,000	Not Cost Effective
AC/DC-28R	*$399,452*	$1,600,000	Not Cost Effective
CB-22R	*$160,164*	$4,600,000	Not Cost Effective
CC-22R	$0	$2,200,000	Not Cost Effective
CW-26R	*$521,496*	$1,500,000	Not Cost Effective
FW-17R	*$583,446*	$2,800,000	Not Cost Effective
OT-08R	*$1,002,584*	$4,400,000	Not Cost Effective

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit"			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
Enhancements Related to AC and DC Power			
AC/DC-08	Increase training on response to loss of 120V AC buses that cause inadvertent actuation signals.	Criterion D Very Low Benefit	Abnormal Procedure DB-OP-2532 addresses the loss of both AC and DC power to both the Non-Nuclear Instrumentation (NNI) and the ICS that are powered from uninterruptible AC instrumentation distribution panels YAU and YBU. It is judged that operator awareness to the required actions is well established. *This SAMA should remain Criterion D.*
AC/DC-16	Improve uninterruptible power supplies.	Criterion D Very Low Benefit	Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to uninterruptible power supplies has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
Enhancements Related to ATWS Events			
AT-01	Add an independent boron injection system.	Criterion D Very Low Benefit	Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to emergency boration has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
AT-02	Add a system of relief valves to prevent equipment damage from pressure spikes during an ATWS.	Criterion D Very Low Benefit	Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to ATWS pressure relief has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
AT-07	Install motor generator set trip breakers in control room.	Criterion D Very Low Benefit	Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to reactor trip has an RRW value above the minimum cost of a hardware modification Also, if the reactor power is not decreasing, procedures instruct the operators to first de-energize substations E2 and F2, and, if necessary, locally open reactor trip breakers in the Low Voltage Switchgear room. This SAMA should remain Criterion D.
Enhancements Related to Containment Bypass			
CB-02	Add redundant and diverse limit switches to each CIV.	Criterion D Very Low Benefit	Failure of containment isolation typically leads to a LERF if core damage has occurred. LERF results are dominated by containment bypass events such as SGTR and ISLOCA events. Containment isolation is not shown to be a significant contributor to LERF in the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CB-03	Increase leak testing of valves in ISLOCA paths.	Criterion D Very Low Benefit	HPI and LPI injection check valves are leak tested per Appendix J. DHR suction lines are not tested, but rather than a leakage test, it is judged that continuously monitoring these valves at power would be preferable to leakage test. A SAMA candidate to continuously monitor the DHR suction valves is provided in SAMA candidate CB-21. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CB-04	Install self-actuating CIVs.	Criterion D Very Low Benefit	Important CIVs receive a close signal from the safety actuation system. Many are air-operated and fail in the closed position. It is judged that self-actuating valves would not provide any significant increase in the reliability of isolation. Containment isolation is not shown to be a significant contributor to CDF or LERF in the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CB-06	Ensure ISLOCA releases are scrubbed. One method is to plug drains in potential break areas so that break point will be covered with water.	Criterion D Very Low Benefit	This SAMA candidate would have very little benefit. It is likely that the break would be well above floor drain level. Therefore, a significant height of water would be required before any scrubbing took place. At these levels, the water level would likely have undesirable effects, such as threatening mitigating equipment due to flooding. This conclusion remains valid for the 95% CDF uncertainty results. *This SAMA should remain Criterion D.*
CB-09	Institute a maintenance practice to perform a 100% inspection of steam generator tubes during each refueling outage.	Criterion D Very Low Benefit	*Davis-Besse is scheduled to replace the steam generators in ~~2013~~ 2014, which would result in inspecting new steam generator tubes.* Therefore, this SAMA candidate is considered very low benefit for Davis-Besse. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CB-18	Direct steam generator flooding after a SGTR, prior to core damage.	Criterion D Very Low Benefit	Flooding the SG prior to core damage could impact efforts to mitigate the SGTR. For example, flooding may present a risk to the operation of the TDAFW pumps by risking steam generator overfill. *Disposition of this SAMA candidate is addressed in the response to RAI 5.i.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CB-19	Vent MSSVs in containment.	Criterion D Very Low Benefit	This SAMA candidate would result in plant decay heat being deposited into primary containment, resulting in a harsh environment. The possible advantages for SGTR will be offset by the negative impacts for other events where secondary steam is deposited into containment with intact steam generators. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CB-20	Install relief valves in the CCW system.	Criterion D Very Low Benefit	Based on the top 100 cutsets and component basic event importance, ISLOCA in the CCW is not significant risk contributor at Davis-Besse. An ISLOCA occurring in the CCW system is not a risk contributor in the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
Enhancements Related to Core Cooling Systems			
CC-11	Modify procedures to throttle LPI pumps earlier in medium or large break LOCAs to maintain BWST inventory.	Criterion D Very Low Benefit	Davis-Besse operators are prohibited from throttling LPI pumps earlier in medium or large break LOCAs to maintain BWST inventory. If BWST flow was throttled down to reduce flowrate, the additional time gained is approximately 20 minutes, which, from a PRA perspective, is of low benefit for a LOCA condition. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CC-13	Upgrade the chemical and volume control system to mitigate small break LOCAs.	Criterion D Very Low Benefit	The make-up system can be used to provide make-up to the RCS in the event of a small LOCA. Because of the separate HPI and make-up systems, the plant has essentially four separate systems capable of injecting from the BWST into the RCS at high pressure. This was identified as a unique safety feature in the IPE. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CC-21	Reduce the BWST level at which switchover to containment recirculation is initiated.	Criterion D Very Low Benefit	Reducing the level at which switchover occurs (nine feet) would not significantly extend the time to switchover, and would increase the probability of pump failure due to loss of suction head. Davis-Besse has installed more accurate BWST level instrumentation that allows reaching a lower level prior to switchover to recirculation. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
Enhancements Related to Containment Phenomena			
CP-03	Use the fire water system as a backup source for the containment spray system.	Criterion D Very Low Benefit	Davis-Besse has a very large dry containment. Containment over-pressurization is not a significant risk contributor. This conclusion remains valid for the 95% LERF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CP-06	Enhance fire protection system hardware and procedures.	Criterion D Very Low Benefit	This SAMA candidate addresses the scrubbing of radioactive releases into certain areas by actuating the fire protection system. Although some scrubbing benefits might be realized, this SAMA candidate presents the risk of impacting required equipment by spray or flooding. This could only be performed with fire protection systems that could be remotely actuated. If the temperature in certain areas became high enough, some existing fire protection systems may automatically actuate. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CP-16	Delay containment spray actuation after a large break LOCA.	Criterion D Very Low Benefit	The delay time that could be realized if containment spray was delayed would be less than 10 minutes. This SAMA candidate is considered to be of very low benefit. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CP-17	Install automatic containment spray pump header throttle valves.	Criterion D Very Low Benefit	The capability already exists at Davis-Besse to throttle containment spray after the switchover to the sump. The delay time that could be realized if containment spray was throttled would be less than 10 minutes. This SAMA candidate is considered to be of very low benefit. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CP-19	Install a redundant containment fan system.	Criterion D Very Low Benefit	Based on component basic event importance, containment fan coolers are not significant risk contributors at Davis-Besse. This SAMA candidate is considered to be very low benefit. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CP-20	Install or use an independent power supply to the hydrogen control system using either new batteries, a non-safety grade portable generator, existing station batteries, or existing AC/DC independent power supplies, such as the security system diesel generator.	Criterion D Very Low Benefit	Davis-Besse has a very large dry containment. Hydrogen burn does not present a significant risk in terms of LERF. This SAMA candidate is considered to be very low benefit. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CP-21	Install a passive hydrogen control system.	Criterion D Very Low Benefit	This SAMA would mitigate large early releases resulting from a hydrogen burn. LERF is dominated by containment bypass events such as SGTR and ISLOCA. Failure of containment is not a significant contributor to LERF. This SAMA candidate is considered to be very low benefit. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
Enhancements Related to Cooling Water			
CW-01	Add redundant DC control power for service water pumps.	Criterion D Very Low Benefit	Failure of DC power would impact much more than service water and improving the reliability of DC power to only service water would have very limited value. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to service water performance has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
CW-04	Add a redundant service water pump.	Criterion D Very Low Benefit	Davis-Besse has three service water pumps. In addition, the normally running cooling tower makeup pump is the preferred supply of service water following loss of service water. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to service water performance has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CW-05	Enhance the screen wash system.	Criterion D Very Low Benefit	The Davis-Besse water supply from Lake Erie travels through a long canal before reaching the intake structure. There is a screen at the intake from Lake Erie. The long distance traveled through the canal results in a significant fraction of material passing through the initial screen settling out prior to reaching the intake structure. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to service water performance has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
CW-06	Cap downstream piping of normally closed CCW drain and vent valves.	Criterion D Very Low Benefit	Loss of CCW through drain and vent lines is not considered to be a significant contributor to loss of CCW. These lines are small, and any leakage would likely be low. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CW-08	Enhance loss of CCW procedure to underscore the desirability of cooling down the RCS prior to seal LOCA.	Criterion D Very Low Benefit	Seal LOCA is not a concern at Davis-Besse if the RCPs are tripped. On loss of CCW, the makeup pumps can continue operation for at least one hour. Therefore, if operators trip the RCPs within one hour of loss of CCW, an RCP seal LOCA is not a risk concern. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CW-09	Additional training on loss of CCW.	Criterion D Very Low Benefit	Seal LOCA is not a concern at Davis-Besse if the RCPs are tripped. On loss of CCW, the makeup pumps can continue operation for at least one hour. Therefore, if operators trip the RCPs within one hour of loss of CCW, an RCP seal LOCA is not a risk concern. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CW-12	Increase charging pump lube oil capacity.	Criterion D Very Low Benefit	Davis-Besse makeup pumps can operate for at least one hour on loss of CCW. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to charging (make-up) pump performance has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
CW-15	Use existing hydro test pump for RCP seal injection.	Criterion D Very Low Benefit	Seal LOCA is not a concern at Davis-Besse if the RCPs are tripped. On loss of CCW, the makeup pumps can continue operation for at least one hour. Therefore, if operators trip the RCPs within one hour of loss of CCW, an RCP seal LOCA is not a risk concern. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CW-18	Prevent make-up pump flow diversion through the relief valves.	Criterion D Very Low Benefit	The make-up system is continuously operating. Malfunctions of relief valves would be immediately detected during operation and corrected. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related make-up flow diversion has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
Enhancements Related to Internal Flooding			
FL-01	Improve inspection of rubber expansion joints on main condenser.	*Revised to read:* Criterion F Considered for Further Evaluation	A large circulating water flood in the turbine building has associated basic event FL7 that is above the minimum cost of a procedure change (although less that a hardware modification). This SAMA candidate will be considered for further evaluation.
Enhancements Related to Fire Risk			
FR-01	Replace mercury switches in fire protection system.	Criterion D Very Low Benefit	Inadvertent actuation of fire protection water is not considered risk significant and currently not modeled in the PRA. Any fire protection system water should be handled by existing drains and is not considered a significant flooding threat. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
FR-02	Upgrade fire compartment barriers.	Criterion D Very Low Benefit	The Davis-Besse IPEEE did not identify any weakness in the fire barrier performance. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
FR-03	Install additional transfer and isolation switches.	Criterion D Very Low Benefit	Currently, isolation switches exist for a control evacuation. Some manual actions beyond operation of isolation switches are required (e.g., plugging connectors, removing/inserting fuse blocks). Adding additional transfer/isolation switches is not considered to be of significant benefit. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
FR-04	Enhance fire brigade awareness.	Criterion D Very Low Benefit	The Davis-Besse IPEEE did not identify any weakness in fire brigade performance. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
FR-05	Enhance control of combustibles and ignition sources.	Criterion D Very Low Benefit	The Davis-Besse IPEEE did not identify any weakness in the combustible control program. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
Enhancements Related to Feedwater and Condensate			
FW-03	Install an independent diesel for the CST make-up pumps.	Criterion D Very Low Benefit	Davis-Besse has the capability of replenishing the CST using fire protection water. This can be done even on loss of AC power. Adding diesel for condensate makeup pumps would add little benefit. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
FW-05	Install manual isolation valves around the TDAFW pump steam admission valves.	Criterion D Very Low Benefit	The purpose of the SAMA candidate was to reduce dual turbine-driven pump maintenance unavailability. Although manual isolation valves do not exist, Davis-Besse has valves within the steam lines that allow isolation of one TDAFW pump for maintenance while leaving the second TDAFW pump available. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
FW-07	Install a new condensate storage tank (AFW storage tank).	Criterion D Very Low Benefit	Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to CST performance has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
FW-12	Change failure position of condenser make-up valve if the condenser make-up valve fails open on loss of air or power.	Criterion D Very Low Benefit	On loss of air or electric power, several components required for secondary heat removal would be lost; therefore the state of the condenser make-up valve is not relevant. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
FW-15	Replace existing pilot-operated relief valves with larger ones, such that only one is required for successful feed and bleed.	Criterion D Very Low Benefit	Failure of the PORV to open only shows up in the Level 1 PRA importance measures with a RRW of 1.006 (cutoff 1.005). It does not show up in the top cutsets or the LERF importance list. Therefore, it is judged to be very low benefit. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to PORV opening or capacity has an RRW value above the minimum cost of a hardware modification *This SAMA should remain Criterion D.*
Enhancements Related to Heating, Ventilation and Air Conditioning (HVAC)			
HV-04	Add a switchgear room high temperature alarm.	Criterion D Very Low Benefit	The high voltage switchgear rooms do not require forced ventilation. Low voltage switchgear rooms require forced ventilation. Operators monitor the temperature of the low voltage switchgear rooms during their plant tours. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to switchgear ventilation has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
HV-05	Create ability to switch emergency feedwater room fan power supply to station batteries in an SBO.	Criterion D Very Low Benefit	Loss of ventilation to AFW is not a risk significant contributor at Davis-Besse. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
HV-06	Provide procedural guidance for establishing an alternate means of room ventilation to the service water pump room.	Criterion D Very Low Benefit	Service water ventilation includes four 50% fans. Loss of service water ventilation is not a significant risk contributor at Davis-Besse. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to service water room ventilation has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
Enhancements Related to Instrument Air and Nitrogen Supply			
IA-02	Modify procedure to provide ability to align diesel power to more air compressors.	Criterion D Very Low Benefit	Service Air and Instrument Air are not significant risk contributors based on top cutsets and risk importance measures. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to air compressors has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
IA-03	Replace service and instrument air compressors with more reliable compressors that have self-contained air cooling by shaft-driven fans.	Criterion D Very Low Benefit	Service Air and Instrument Air are not significant risk contributors based on top cutsets and risk importance measures. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to service or instrument air compressors has an RRW value above the minimum cost of a hardware modification *This SAMA should remain Criterion D.*
Enhancements Related to Seismic Risk			
SR-01	Increase seismic ruggedness of plant components.	Criterion D Very Low Benefit	The Seismic Qualifications Utility Group (SQUG) previously identified the need for additional seismic restraints in the plant. These restraints have already been added. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
SR-02	Provide additional restraints for CO_2 tanks.	Criterion D Very Low Benefit	The CO_2 tanks are located outdoors. These tanks supply only the turbine generator. No other components are protected with CO_2. A seismic failure of the CO_2 tanks has minimal risk. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
Other Enhancements			
OT-01	Install digital large break LOCA protection system.	Criterion D Very Low Benefit	Large break LOCA is not a significant risk contributor (0.2% CDF). Davis-Besse has a Containment Leakage Detection System (FLUS) to identify leaks from vessel penetrations and nozzles. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
OT-04	Improve maintenance procedures.	Criterion D Very Low Benefit	Davis-Besse has a qualified Maintenance Rule program in place. No deficiencies in maintenance practices have been identified. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
OT-05	Increase training and operating experience feedback to improve operator response.	*Revised to read:* Criterion B Already Implemented	FENOC provides PRA information, such as risk-significant initiating events, high worth operator actions and high worth equipment, to various departments, including Operations Training, and presents this information on posters throughout the plant.

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
OT-07	Install secondary side guard pipes up to the MSIVs.	Criterion D Very Low Benefit	Steam line breaks are not a significant contributor to CDF or LERF based on top cutsets or basic event importance. The derived benefit would not justify the implementation cost required. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to main steam breaks has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*

Item 7

For certain SAMAs considered in the ER, there may be lower-cost alternatives that could achieve much of the risk reduction at a lower cost. In this regard, discuss whether any lower-cost alternatives to those Phase II SAMAs considered in the ER would be viable and potentially cost-beneficial. Evaluate the following SAMAs (previously found to be potentially cost-beneficial at other Babcock and Wilcox plants), or indicate if the particular SAMA has already been considered. If the latter, indicate whether the SAMA has been implemented or has been determined to not be cost-beneficial at Davis-Besse Nuclear Power Station.

Question RAI 7.a

Automate reactor coolant pump trip on high motor bearing cooling temperature.

RESPONSE RAI 7.a

[The response to RAI 7.a is edited as shown in the text and Tables 7.a-1, 7.a-2 and 7.a-3.]
[NOTE: * One sensitivity case no longer applies and has been deleted; see the revised response to RAI 6.i.ii]

A SAMA candidate (CW-26R) to provide an automatic reactor coolant pump trip on loss of cooling to the RCP seal thermal barrier cooler and loss of seal injection flow was evaluated for Davis-Besse. Table 7.a-1 and Table 7.a-2, below, provide the internal event and total benefit results for SAMA candidate CW-26R, respectively. *Table 7.a-3, below, provides the final results for the ~~ten~~ nine* [*] *sensitivity cases for SAMA candidate CW-26R.* The implementation cost for this SAMA candidate was estimated as $1,500,000. Therefore, this SAMA candidate is not cost-beneficial at Davis-Besse.

Table 7.a-1: Internal Events Benefit Results for SAMA Candidate CW-26R

Case	CW-26R (Auto_RCP)
Off-site Annual Dose (rem)	*2.05E+00*
Off-site Annual Property Loss ($)	*3.49E+03*
Comparison CDF	1.0E-05
Comparison Dose (rem)	*2.12E+00*
Comparison Cost ($)	*3.59E+03*
Enhanced CDF	7.7E-06
Reduction in CDF	**23.00%**
Reduction in Off-site Dose	***3.30%***
Immediate Dose Savings (On-site)	$186
Long Term Dose Savings (On-site)	$812
Total Accident Related Occupational Exposure **(AOE)**	$998
Cleanup/Decontamination Savings (On-site)	$30,443
Replacement Power Savings (On-site)	$30,801
Averted Costs of On-site Property Damage **(AOSC)**	$61,244
Total On-site Benefit	**$62,242**
Averted Public Exposure **(APE)**	*$1,718*
Averted Off-site Damage Savings **(AOC)**	*$1,227*
Total Off-site Benefit	***$2,945***
Total Benefit (On-site + Off-site)	***$65,187***

Table 7.a-2: Total Benefit Result for SAMA Candidate CW-26R

	CW-26R (Auto_RCP)
Internal Events	*$65,187*
Fires, Seismic, Other	*$299,860*
Total Benefit	*$365,047*

Table 7.a-3: Final Results of the Sensitivity Cases for SAMA Candidate CW-26R

SAMA ID	Repair Case	Low Discount Rate Case	High Discount Rate Case	On-site Dose Case	On-site Clean-up Case
CW-26R	*$227,059*	*$551,324*	*$251,436*	*$370,702*	*$421,875*

SAMA ID	Replacement Power Case	Multiplier Case	Evacuation Speed	95th CDF Case
CW-26R	*$480,823*	*$521,496*	*$365,047*	*$529,319*

Question RAI 7.d

Automate refill of the borated water storage tank (BWST).

RESPONSE RAI 7.d

[The response to RAI 7.d is edited as shown in the text and Tables 7.d-1 and 7.d-3.] [NOTE: * One sensitivity case no longer applies and has been deleted; see the revised response to RAI 6.i.ii]

A SAMA candidate (CC-22R) to provide an automatic refill of the borated water storage tank was evaluated for Davis-Besse. Table 7.d-1 and Table 7.d-2, below, provide the internal event and total benefit results for SAMA candidate CC-22R, respectively. *Table 7.d-3, below, provides the final results for the ~~ten~~ nine* [*] *sensitivity cases for SAMA candidate CC-22R.* The implementation cost for this SAMA candidate was estimated as $2,200,000. Therefore, this SAMA candidate is not cost-beneficial at Davis-Besse.

Table 7.d-1: Internal Events Benefit Results for SAMA Candidate CC-22R

Case	CC-22R (Auto_BWST)
Off-site Annual Dose (rem)	*2.12E+00*
Off-site Annual Property Loss ($)	*3.59E+03*
Comparison CDF	1.0E-05
Comparison Dose (rem)	*2.12E+00*
Comparison Cost ($)	*3.59E+03*
Enhanced CDF	1.0E-05
Reduction in CDF	**0.00%**
Reduction in Off-site Dose	**0.00%**
Immediate Dose Savings (On-site)	$0
Long Term Dose Savings (On-site)	$0
Total Accident Related Occupational Exposure **(AOE)**	$0
Cleanup/Decontamination Savings (On-site)	$0
Replacement Power Savings (On-site)	$0
Averted Costs of On-site Property Damage **(AOSC)**	$0
Total On-site Benefit	**$0**
Averted Public Exposure **(APE)**	$0
Averted Off-site Damage Savings **(AOC)**	$0
Total Off-site Benefit	**$0**
Total Benefit (On-site + Off-site)	**$0**

Table 7.d-2: Total Benefit Result for SAMA Candidate CC-22R

	CC-22R (Auto_BWST)
Internal Events	$0
Fires, Seismic, Other	$0
Total Benefit	$0

Table 7.d-3: Final Results of the Sensitivity Cases for SAMA Candidate CC-22R

SAMA ID	Repair Case	Low Discount Rate Case	High Discount Rate Case	On-site Dose Case	On-site Clean-up Case
CC-22R	$0	$0	$0	$0	$0

SAMA ID	Replacement Power Case	Multiplier Case	Evacuation Speed	95th CDF Case
CC-22R	$0	$0	*$0*	$0

Question RAI 7.e

Automate start of auxiliary feedwater (AFW) pump in the event the automated emergency feedwater (EFW) system is unavailable.

RESPONSE RAI 7.e

[The response to RAI 7.e is edited as shown in the text and Tables 7.e-1, 7.e-2 and 7.e-3.]
[NOTE: * One sensitivity case no longer applies and has been deleted; see the revised response to RAI 6.i.ii]

A SAMA candidate (FW-17R) to automatically start the auxiliary feedwater pump when the emergency feedwater system is unavailable was evaluated for Davis-Besse. Based on the Davis-Besse design, this SAMA was interpreted as automatically starting the motor driven feedwater pump in the event both turbine-driven auxiliary feedwater pumps were not available. Table 7.e-1 and Table 7.e-2, below, provide the internal event and total benefit results for SAMA candidate FW-17R, respectively. *Table 7.e-3, below, provides the final results for the ~~ten~~ nine* [*] *sensitivity cases for SAMA candidate FW-17R.* The implementation cost for this SAMA candidate was estimated as $2,800,000. Therefore, this SAMA candidate is not cost-beneficial at Davis-Besse.

Table 7.e-1: Internal Events Benefit Results for SAMA Candidate FW-17R

Case	FW-17R (Auto_MDFP)
Off-site Annual Dose (rem)	*2.00E+00*
Off-site Annual Property Loss ($)	*3.40E+03*
Comparison CDF	1.0E-05
Comparison Dose (rem)	*2.12E+00*
Comparison Cost ($)	*3.59E+03*
Enhanced CDF	7.5E-06
Reduction in CDF	**25.00%**
Reduction in Off-site Dose	***5.66%***
Immediate Dose Savings (On-site)	$202
Long Term Dose Savings (On-site)	$882
Total Accident Related Occupational Exposure **(AOE)**	$1,085
Cleanup/Decontamination Savings (On-site)	$33,091
Replacement Power Savings (On-site)	$33,479
Averted Costs of On-site Property Damage **(AOSC)**	$66,570
Total On-site Benefit	**$67,655**
Averted Public Exposure **(APE)**	$2,945
Averted Off-site Damage Savings **(AOC)**	*$2,331*
Total Off-site Benefit	***$5,276***
Total Benefit (On-site + Off-site)	***$72,931***

Table 7.e-2: Total Benefit Result for SAMA Candidate FW-17R

	FW-17R (Auto_MDFP)
Internal Events	*$72,931*
Fires, Seismic, Other	*$335,481*
Total Benefit	*$408,412*

Table 7.e-3: Final Results of the Sensitivity Cases for SAMA Candidate FW-17R

SAMA ID	Repair Case	Low Discount Rate Case	High Discount Rate Case	On-site Dose Case	On-site Clean-up Case
FW-17R	*$258,425*	*$617,207*	*$282,195*	*$414,559*	*$470,181*

SAMA ID	Replacement Power Case	Multiplier Case	Evacuation Speed	95th CDF Case
FW-17R	*$534,255*	*$583,446*	*$408,412*	*$592,197*

Question RAI 7.f

Purchase or manufacture of a "gagging device" that could be used to close a stuck-open steam generator safety valve for a SGTR event prior to core damage.

RESPONSE RAI 7.f

[The response to RAI 7.f is edited as shown in the text and Tables 7.f-1, 7.f-2 and 7.f-3.] [NOTE: * One sensitivity case no longer applies and has been deleted; see the revised response to RAI 6.i.ii]

A SAMA candidate (CB-22R) to use a "gagging" device that could be used to close a stuck-open steam generator safety valve for a SGTR was evaluated for Davis-Besse. Table 7.f-1 and Table 7.f-2, below, provide the internal event and total benefit results for SAMA candidate CB-22R, respectively. *Table 7.f-3, below, provides the final results for the ~~ten~~ nine* [*] *sensitivity cases for SAMA candidate CB-22R.* The implementation cost for this SAMA candidate was estimated as $4,600,000. The high implementation cost of this SAMA candidate is based on replacement of the safety valves with a new design that includes a gagging feature. Therefore, this SAMA candidate is not cost-beneficial at Davis-Besse.

Table 7.f-1: Internal Events Benefit Results for SAMA Candidate CB-22R

Case	CB-22R (Gagging_Device)
Off-site Annual Dose (rem)	*1.86E+00*
Off-site Annual Property Loss ($)	*3.14E+03*
Comparison CDF	1.0E-05
Comparison Dose (rem)	*2.12E+00*
Comparison Cost ($)	*3.59E+03*
Enhanced CDF	9.7E-06
Reduction in CDF	**3.00%**
Reduction in Off-site Dose	***12.26%***
Immediate Dose Savings (On-site)	$24
Long Term Dose Savings (On-site)	$106
Total Accident Related Occupational Exposure **(AOE)**	$130
Cleanup/Decontamination Savings (On-site)	$3,971
Replacement Power Savings (On-site)	$4,018
Averted Costs of On-site Property Damage **(AOSC)**	$7,988
Total On-site Benefit	**$8,119**
Averted Public Exposure **(APE)**	$6,380
Averted Off-site Damage Savings **(AOC)**	*$5,522*
Total Off-site Benefit	***$11,902***
Total Benefit (On-site + Off-site)	***$20,020***

Table 7.f-2: Total Benefit Result for SAMA Candidate CB-22R

	CB-22R (Gagging_Device)
Internal Events	*$20,020*
Fires, Seismic, Other	*$92,094*
Total Benefit	*$112,115*

Table 7.f-3: Final Results of the Sensitivity Cases for SAMA Candidate CB-22R

SAMA ID	Repair Case	Low Discount Rate Case	High Discount Rate Case	On-site Dose Case	On-site Clean-up Case
CB-22R	*$94,116*	*$171,489*	*$82,166*	*$112,852*	*$119,527*

SAMA ID	Replacement Power Case	Multiplier Case	Evacuation Speed	95th CDF Case
CB-22R	*$127,216*	*$160,164*	*$112,115*	*$162,566*

Attachment 3
L-12-244

Review for Impacts to Responses to Requests for Additional Information for the Review of the Davis Besse Nuclear Power Station, Unit No. 1 (Davis-Besse), License Renewal Application, Environmental Report, Attachment E, Severe Accident Mitigation Alternatives Analysis
Submitted by FENOC Letter Dated September 1, 2011 (ML11250A068)

FirstEnergy Nuclear Operating Company (FENOC) performed a review, based on the revised SAMA Analysis, for impacts to the supplemental responses to Nuclear Regulatory Commission (NRC) supplemental requests for additional information (RAIs) for the Severe Accident Mitigation Alternatives (SAMA) Analysis submitted by FENOC letter dated September 1, 2011 (ML11250A068). Based on the changes to the SAMA Analysis, no revision to the FENOC supplemental responses provided in the September 1, 2011, letter is necessary. The list of supplemental RAI s contained in the letter is as follows:

SAMA RAI Supplemental Responses – No Revision	
1.d	7.b
4.b	7.c
5.b	7.d
5.d	7.e
6.j	7.f
7.a	

Enclosure

Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse)

Letter L-12-244

Amendment No. 29 to the
Davis-Besse License Renewal Application

License Renewal Application
Environmental Report (ER) Sections Affected

Environmental Report	Section E.8.6	Table E.3-27
Section 4.20	Section E.9	Table E.3-28
		Table E.3-29
ER Attachment E	**Section E.10**	Table E.3-30
Executive Summary	Table E.3-6	Table E.3-31
Section E.3.1.2.4	Table E.3-11	Table E.3-32
Section E.3.4.1	Table E.3-13	Table E.3-33
Section E.3.4.8	Table E.3-18	Table E.4-1
Section E.3.5.2.2	Table E.3-19	Table E.5-4
Section E.3.5.2.3	Table E.3-20	Table E.6-1
Section E.3.5.2.4	Table E.3-21	Table E.7-2
Section E.4.1	Table E.3-22	Table E.7-3
Section E.4.2	Table E.3-23	Table E.7-5
Section E.4.5	Table E.3-24	Table E.8-1
Section E.5.6	Table E.3-25	
Section E.7.1.2	Table E.3-26	Section E.11

The amendment to the License Renewal Application (LRA) ER Sections and Tables included in this Enclosure are a result of the revision to the Severe Accident Mitigation Alternatives (SAMA) Analysis based on correction of the five SAMA Analysis errors, unless otherwise indicated. The Enclosure identifies the change to the LRA by Affected LRA Section, LRA Page No., and Affected Paragraph and Sentence. The count for the affected paragraph, sentence, bullet, etc., starts at the beginning of the affected Section or at the top of the affected page, as appropriate. The sentence affected is printed in *italics* with deleted text ~~*lined-out*~~ and added text *underlined*.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section 4.20	**4.20-3 & 4.20-4**	**Final paragraph**

Based on the responses to RAIs 4.b (see FENOC Letter dated June 24, 2011 (ML11180A233) and 5.c, and the revised SAMA Analysis, Environmental Report (ER) Section 4.20, "Severe Accident Mitigation Alternatives," the last bulleted item and final paragraph, are replaced in their entirety to read as follows:

- Sensitivity Analysis – Sensitivity cases were performed to investigate the sensitivity of the results to certain modeling assumptions in the Davis Besse SAMA analysis. Nine sensitivity cases were investigated. These cases examined the impacts of assuming damaged plant equipment is repaired and refurbished following an accident, a lower discount rate, a higher discount rate, higher on-site dose estimates, higher total on-site cleanup costs, higher costs for replacement power, a higher external event hazard groups' multiplier, a reduced evacuation speed, and use of a 95th percentile core damage frequency uncertainty factor for internal events. Details on the sensitivity cases are discussed in Attachment E, Section E.8.

The results of the evaluation of 168 SAMA candidates identified one cost-beneficial enhancement at Davis Besse. Assuming a lower discount rate, higher dose rates, higher onsite clean-up costs, increased replacement power costs, increased external event multiplier, reduced evacuation speed, and use of a 95th percentile core damage frequency uncertainty factor for internal events identified the same SAMA candidate to be cost-beneficial. The SAMA candidate identified in the base case and sensitivity cases is not related to plant aging. Therefore, the identified cost-beneficial SAMA candidate is not a required modification for the license renewal period. Nevertheless, this SAMA candidate will be considered through the normal FENOC processes for evaluating possible modifications to the plant.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Attachment E – Executive Summary	**E-9**	**4th and 5th paragraphs**

Based on the responses to RAIs 4.b (see FENOC Letter dated June 24, 2011 (ML11180A233) and 5.c, and the revised SAMA Analysis, the Executive Summary of ER Attachment E, "Severe Accident Mitigation Alternatives Analysis," paragraphs four and five, are revised to read as follows:

The cost-benefit evaluation of SAMA candidates performed for Davis-Besse provides significant insight into the continued operation of Davis-Besse. *The results of the evaluation of ~~167~~ 168 SAMA candidates indicate ~~no enhancements~~ one enhancement to be cost-beneficial for implementation at Davis-Besse. The cost-beneficial SAMA candidate is AC/DC-03, which adds a portable diesel-driven battery charger to the DC system.*

~~However, the~~ The sensitivity cases performed for this analysis found ~~one~~ the same SAMA candidate (AC/DC-03) to be cost-beneficial for implementation at Davis-Besse under the assumptions of ~~three of the sensitivity cases (lower discount rate, replacement power, and multiplier). SAMA candidate AC/DC-03 considered the addition of a portable diesel-driven battery charger for the DC system.~~ a lower discount rate, higher dose rates, higher onsite clean-up costs, increased replacement power costs, increased external event multiplier, reduced evacuation speed, and use of a 95th percentile core damage frequency uncertainty factor for internal events. While the identified SAMA candidate is not related to plant aging and therefore not required to be resolved as part of the relicensing effort, FENOC will, nonetheless, consider implementation of this candidate through normal processes for evaluating possible changes to the plant.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.3.1.2.4	**E-28**	**Entire Section**

In response to RAI 3.c, ER Section E.3.1.2.4, "External Event Severe Accident Risk," is deleted in its entirety, as follows:

~~*E.3.1.2.4 External Event Severe Accident Risk*~~

~~*This section describes the method used to address external events risk.*~~

~~*As discussed in Section E.3.1.2.2, Davis Besse used the SMA to evaluate the risk from seismic events. While this methodology does not provide a quantitative result, the resolution of outliers ensures that the seismic risk is low and further cost-beneficial seismic improvements are not expected. Also, as discussed in Section E.3.1.2.3, no other external events were found to exceed the screening criteria. Therefore, the FIVE results were used as a measure of total external events risk.*~~

~~*As discussed in Section E.3.1.2.1, using the EPRI FIVE methodology, Davis Besse conservatively estimated the Fire CDF to be 2.5E-05/yr. Since the FIVE methodology contains numerous conservatisms, a more realistic assessment could result in a substantially lower fire CDF. As noted in NEI 05-01 (Reference 2), the NRC staff has accepted that a more realistic fire CDF may be a factor of three less than the screening value obtained from a FIVE analysis.*~~

~~*Based on the Davis Besse FIVE CDF of 2.5E-05/yr, a factor of three reduction would result in a fire CDF of approximately 8.3E-06/yr. This value is the same order of magnitude as the internal events CDF of 9.2E-06/yr. Therefore, this justifies use of an external events multiplier of three to the averted cost estimates (for internal events) to represent the additional SAMA benefits in external events.*~~

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.3.4.1	**E-33 & E-34**	**Last paragraph and bulleted list**

ER Section E.3.4.1, "Introduction," last paragraph and bulleted list, are revised to read as follows:

The Level 3 PRA analysis considered a base case and ~~eleven~~ ten sensitivity cases to account for variation in data and assumptions. The following list describes the sensitivity cases, which are discussed in Section ~~E.8~~ E.3.5.2:

- Case S1 – Use estimated 2060 site population data (with an escalation rate of 4.7%/decade); the same escalation rate for the base case population to 2040
- Case S2 – Use a less conservative escalation rate of 1.5% to estimate the 50-mile population around Davis Besse in 2040
- Case S3 – Set all watershed indices to "1"
- Case M1 – Use 2007 meteorological data
- ~~*Case M2 – Use meteorological data from circa late 1990s*~~
- Case A1 – Use an alternative method to estimate PLHEAT
- Case A2 – Use more extreme meteorological boundary conditions
- Case A3 – Use a longer OALARM value to better reflect operator's ability to react
- *Case E1 – Use a more realistic (~~higher~~ faster) evacuation speed ~~of evaluation~~ (ESPEED)*
- Case E2 – Set sheltering shielding factors based on brick house (versus wood housing used in the base case)
- *Case E3 – Use a slower evacuation speed (ESPEED)*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.3.4.8	**E-41**	**RELCST (Relocation Cost), 2nd and 4th paragraphs**

ER Section E.3.4.8, "Economic Data," "RELCST" subsection, 2nd and 4th paragraphs, are revised to read as follows:

RELCST was estimated using the evacuation costs plus the average property cost per person. The average property cost per person was calculated from the total property value in the state, which can be found on the individual state's Department of Revenue websites:

- *$256,088,369,000 for Ohio (Reference 25, Table PD-30);*
  Ohio property values obtained were 'tax assessment' values, which are 35% of total property value, so the Ohio property value needs to be corrected by dividing by 0.35 to obtain total property value
- $340,545,761,049 for Michigan (Reference 26, Exhibit 22)

The total property cost was divided by the total population (11,353,140 for Ohio and 9,938,444 for Michigan) (Reference 27).

For Ohio State, RELCST is ~~$266.34/person-day~~ $381.11/person-day; for Michigan State, RELCST is $310.61/person-day. The average of the Ohio and Michigan RELCST values was used as input in the CHRONC file.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.3.5.2.2	**E-43 & E-44**	**Case M2 subsection – entire subsection**

ER Section E.3.5.2.2, "Meteorological," "Case M2" subsection, is deleted in its entirety, as follows:

~~***Case M2*** *– An additional sensitivity case was performed to further demonstrate the typical nature of any particular year's worth of meteorological data. These data are circa late-1990s, but no specific year could be identified, and therefore are only to be used as a second sensitivity case.*~~

~~*The results in Table E.3-27 are similar to sensitivity case M1, with some minor variability in the consequence, demonstrating the representativeness of any year's worth of meteorological data.*~~

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.3.5.2.3	**E-44**	**Case A2 subsection, 3rd and 4th sentences**

ER Section E.3.5.2.3, "ATMOS," "Case A2" subsection, 3rd and 4th sentences, are revised to read as follows:

Case M2 – A sensitivity case was run with more extreme values of the meteorological boundary parameters, i.e., mixing height (BNDMXH), stability class (IBDSTB), rain rate (BNDRAN), wind speed (BNDWND). In general, the sensitivity case considered all of these boundary parameters collectively (i.e., all considered in one case). *The rain rate boundary condition was set at ~~0.0 mm/hour for the base case; there is no value more conservative than that~~ 30.73 mm/hour (the maximum rainfall in any hour) as a sensitivity case against the base case value of 0.0 mm/hr. The ~~conservative~~ more extreme boundary parameters had no impact on the results as shown in Table E.3-29.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.3.5.2.4	**E-45**	**New subsection**

Based on the response to RAI 4.d, and the revised SAMA Analysis, ER Section E.3.5.2.4, "Early," previously revised by FENOC letter dated June 24, 2011 (ML11180A233), to include a new paragraph for sensitivity case E3 at the end of the section, is revised to read as follows:

Case E3 – The base case was performed with an evacuation speed of 0.58 meters/second, based on Davis-Besse-specific evaluation information, without any correction factor to account for the escalated population. In response to an NRC request for additional information, this sensitivity case was performed to gauge the sensitivity of reducing the evacuation speed. As the population was increased 4.7 percent per decade for the 20 years of license renewal (total increase of 9.6 percent), it was assumed for this sensitivity case that the increase in population was directly proportional to the decrease in evacuation speed. The evacuation speed for this sensitivity is a 9.6 percent decrease from the base case, i.e., 0.52 meters/second. *This change resulted in ~~a minor~~ no increase in ~~the consequence values~~ population dose risk, as shown in Table E.3-33, and only a minor increase in the other consequence values. ~~This is~~ These results are expected as slower evacuation should remove the population from the radiological damage less quickly.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.4.1	**E-47**	**1st paragraph on page**

ER Section E.4.1, "Off-site Exposure Cost," the first paragraph on page E-47, previously revised by FENOC Letter dated June 24, 2011 (ML11180A233), is revised to read as follows:

Table E.3-21 provides the off-site dose for each release category obtained for the base case of the Davis Besse Level 3 PRA weighted by the release category frequency. *The total off-site dose for internal events (D_t) was estimated to be ~~2.30~~ 2.12 person-rem/year.* The APE cost was determined using Equation E.4-2 (Reference 1, Section 5.7.1).

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.4.1	**E-48**	**Equations E.4-6 and E.4-7**

ER Section E.4.1, "Off-site Exposure Cost," equations E.4-6 and E.4-7, previously revised by FENOC Letter dated June 24, 2011 (ML11180A233), are replaced in their entirety to read as follows:

$$Z_{pha} = \left(2{,}000\,\frac{\$}{person-rem}\right)\left(2.12\,\frac{person-rem}{yr}\right) = \$4240/yr \qquad \underline{(E.4\text{-}6)}$$

where,

R = $2,000/person-rem

D_t = 2.12 person-rem/year

The values for the best estimate case are:

C = 12.27 yr

Z_{pha} = $4,240/yr

$$APE = (12.27yr)\left(\frac{\$4240}{yr}\right) = \$52{,}025 \qquad \underline{(E.4\text{-}7)}$$

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.4.2	**E-49**	**1st paragraph, 4th sentence, and equations E.4-8 and E.4-9**

ER Section E.4.2, "Off-site Economic Cost," the first paragraph, fourth sentence and equations E.4-8 and E.4-9, previously revised by FENOC Letter dated June 24, 2011 (ML11180A233), are revised to read as follows:

The term used for off-site economic cost is designated as averted off-site property damage costs (AOCs). The off-site economic loss for a 50-mile radius of the site was determined using the MACCS2 model developed for the Davis Besse Level 3 PRA in Section E.3.4. Table E.3-21 provides the economic loss for each release category obtained for the base case of the Level 3 PRA weighted by the release category frequency. *The total economic loss from internal events (I_t) was estimated to be ~~$1,800~~ $3,590 ($3.59E+03) per year.* The averted cost was determined using Equation E.4-8 from Reference (1), Section 5.7.5.

$$AOC = (C)(I_t) \qquad \text{(E.4-8)}$$

where,

AOC = off-site economic costs associated with a severe accident ($)

C = present value factor (yr)

I_t = monetary value of economic loss per year from internal events before discounting ($/yr)

The values for the base case are:

C = 12.27 yr

I_t = ~~$1,800/yr~~ $3,590/yr

$$AOC = (12.27\text{yr})\left(3{,}590\,\frac{\$}{\text{yr}}\right) = \$44{,}049 \qquad \text{(E.4-9)}$$

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.4.5	**E-55**	**Entire section, including equations**

Based on the responses to RAIs 3.c and 4.b (see FENOC letter dated June 24, 2011 (ML11180A233), and the revised SAMA Analysis, ER Section E.4.5, "Total Cost of Severe Accident Risk," is revised to read:

The total cost of severe accident impact for internal events was calculated by summing the public exposure cost, off-site property damage cost, occupational exposure cost, and on-site economic cost. *The cost of the impact of a severe accident for internal events was ~~$339,331~~ $366,693 as shown in Table E.4-1. Davis Besse does not have external events (i.e., fire, seismic and other external events) PRA from which risk contributors could be combined with the internal events risk. ~~This analysis assumed that the benefit from each hazard group's (i.e., fire, seismic, and other external events) contribution is equivalent to that of internal events. This approach is conservative, based on the discussion in Section E.3.1.2. Therefore, the cost of SAMA candidate implementation was compared with a benefit value of four times (i.e., 1x for internal events plus 3x for external events) that calculated for internal events to include the contribution from internal events, fire, seismic, and other hazard groups.~~ Based on the NRC staff's best estimate, the fire CDF for Davis-Besse is 2.9x10-5/yr [39]. To account for the risk contribution from the fire hazard, a ratio between the fire CDF and internal events CDF was used to determine a fire multiplier of 2.90 (see equation E.4-24).*

$$\frac{\text{Fire CDF}}{\text{Internal Events CDF}} = \frac{2.9\text{x}10^{-5}/\text{yr}}{1.0\text{x}10^{-5}/\text{yr}} = 2.90 \qquad \textit{(E.4-24)}$$

Based on updated probabilistic seismic hazard estimates due to Generic Issue 199, the NRC staff estimated a "weakest link model" seismic CDF for Davis-Besse of $6.7\text{x}10^{-6}$/yr [40]. To account for the risk contribution from the seismic hazard, a ratio between the seismic CDF and internal events CDF was used to determine a seismic multiplier of 0.67 (see equation E.4-25).

$$\frac{\text{Seismic CDF}}{\text{Internal Events CDF}} = \frac{6.7\text{x}10^{-6}/\text{yr}}{1.0\text{x}10^{-5}/\text{yr}} = 0.67 \qquad \textit{(E.4-25)}$$

This analysis conservatively assumed that the benefit from other hazard groups contribution is equivalent to that of internal events. Therefore, the other hazard groups multiplier is 1.0.

To determine the multiplier to account for fire, seismic, and other hazard groups, the individual multipliers are summed; the resulting multiplier is 4.6.

This approach provided a comparison of the cost to the risk reduction estimated for internal and external events for each SAMA candidate. *The maximum benefit for Davis Besse was ~~$1,357,324~~ $2,053.481 as shown in Table E.4-1.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.5.6	**E-63**	**1st sentence**

In response to RAIs 4.b (see FENOC letter dated June 24, 2011 (ML11180A233) and 5.c, ER Section E.5.6, "Initial SAMA Candidate List," the first sentence in the section is 2nd revised to read:

Based on the review of the aforementioned sources, an initial list of ~~167~~ 168 SAMA candidates was assembled.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.7.1.2	**E-68**	**2nd paragraph, 6th & 7th sentence**

ER Section E.7.1.2, "Best-Estimate Benefit Calculation," 2nd paragraph, 6th and 7th sentences are revised to read:

For each case, the benefit from internal events and external events (fire, seismic, and other hazard groups) were summed in a worksheet to ~~determine~~ estimate the total benefit of implementing the SAMA candidate. As discussed in Section E.4.5, the fire, seismic, and other hazard group risk contribution was conservatively estimated to be equivalent to ~~three~~ 4.6 times the internal events risk contribution.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.8.6	**E-73**	**Last two bullets and last paragraph**

ER Section E.8.6, "Other Sensitivity Cases," the last two bullets and last paragraph, are revised to read as follows:

- *The fifth sensitivity case investigated the sensitivity of each analysis to the ~~non-internal~~ external events hazard groups' multiplier by assuming a multiplier of ~~five~~ seven.*

- ~~*The sixth sensitivity case investigated the sensitivity of each analysis to the off-site economic cost. This sensitivity case assumed the off-site ecomonic cost was increased by twenty-five percent.*~~ *The sixth sensitivity case assessed the impact of using an uncertainty factor for internal events based on the 95th percentile CDF for internal events. The uncertainty factor used in this sensitivity case was 1.45.*

The results of the sensitivity cases (Repair, On-site Dose, On-site Cleanup, Replacement Power, Multiplier, and ~~Off-site Economic Cost~~ 95th percentile CDF) are summarized in Table E.8 1.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.9	**E-74**	**1st and 2nd paragraphs**

Based on the responses to RAIs 4.b (see FENOC letter dated June 24, 2011 (ML11180A233) and 5.c, and the revised SAMA Analysis, the first and second paragraphs of ER Section E.9, "Conclusions," are revised to read:

The cost-benefit evaluation of SAMA candidates performed for the Davis-Besse license renewal process provided significant insight into the continued operation of Davis-Besse. *The results of the evaluation of ~~167~~ 168 SAMA candidates indicated ~~no enhancements to be potentially~~ one enhancement to be cost-beneficial for implementation at Davis-Besse. The cost-beneficial SAMA candidate is AC/DC-03, which adds a portable diesel-driven battery charger to the DC system.*

~~However, the~~ The sensitivity cases performed for this analysis also found ~~one~~ the same SAMA candidate (AC/DC-03) to be ~~potentially~~ cost-beneficial for implementation at Davis-Besse under the assumptions of the lower discount rate, higher dose rates, higher on-site clean-up costs, increased replacement power costs, increased external event multiplier, reduced evacuation speed, and 95th percentile CDF sensitivity cases. ~~three of the sensitivity cases (low discount rate, replacement power, and multiplier). SAMA candidate AC/DC-03 considered the addition of a portable diesel-driven battery charger for the DC system.~~ While the identified SAMA candidate is not related to plant aging and therefore not a required modification for the license renewal period, FENOC will, nonetheless, consider implementation of this candidate through the normal processes for evaluating possible plant modifications.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.3-6	**E-83**	**Entire Table**

ER Table E.3-6, "Release Severity Source Term Release Fraction," is replaced in its entirety, and reads as follows:

Table E.3-1: Release Severity Source Term Release Fraction

Release Category	Cesium Iodine % Release
2.1	34.40%
3.4	31.10%
3.2	30.30%
2.2	28.90%
5.2	11.60%
5.4	11.10%
7.2	9.93%
6.2	7.16%
1.2	7.05%
1.4	5.95%
8.2	4.84%
1.3	4.42%
1.1	4.04%
6.1	3.12%
7.1	2.34%
7.6	2.16%
4.2	1.96%
6.4	1.91%
7.8	1.76%
5.1	0.83%
5.3	0.73%
3.1	0.63%
4.4	0.62%
3.3	0.46%
7.5	0.21%
6.3	0.20%
7.4	0.03%
7.3	0.02%
4.1	0.01%
4.3	0.01%
9.2	0.00%
7.7	0.00%
8.1	0.00%
9.1	0.00%

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.3-11	**E-86**	**New row**

ER Table E.3-11, "Total (Permanent and Transient) Escalated Population (50-Mile Radius – Davis Besse) for the Year 2040," previously revised by FENOC letter dated June 24, 2011 (ML11180A233), is revised to add a 'Total Population' row at the bottom of the table, and now reads:

Table E.3-11: Total (Permanent and Transient) Escalated Population (50-Mile Radius – Davis-Besse) for the Year 2040

Sector	1 mile	2 miles	3 miles	4 miles	5 miles	10 miles	20 miles	30 miles	40 miles	50 miles
N	0	0	0	0	0	0	0	0	151518	448232
NNE	6	0	0	0	0	0	0	0	154651	193313
NE	0	0	0	0	0	0	0	0	38663	96657
ENE	0	0	0	0	0	0	828	0	0	0
E	0	0	0	0	0	0	2229	219	0	13561
ESE	0	0	320	0	0	0	11198	50152	20763	104445
SE	662	661	0	0	6786	27558	7443	9301	35612	11828
SSE	661	729	60	71	109	1593	2075	23880	6229	20419
S	4	12	55	328	651	1680	34083	7301	34694	7138
SSW	17	5	82	79	482	5743	4141	6025	26881	12565
SW	37	20	20	469	197	1728	9970	9130	7669	64607
WSW	0	50	0	35	84	1050	8246	12404	47735	14163
W	0	53	72	66	87	847	19318	259606	102087	25871
WNW	683	723	156	0	7274	4821	7009	207932	58896	13460
NW	0	165	595	0	0	1763	0	53092	20356	25771
NNW	20	138	0	0	0	0	0	20080	77289	233548
Total Population										***2,909,792***

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.3-13	**E-87 thru E-93**	**Entire Table**

ER Table E.3-13, "MAAP Output for MACCS2," is replaced in its entirety, to read as follows:

Table E.3-13: MAAP Output for MACCS2

Davis-Besse MAAP Case ID		ST11_RIYVXINN_52Y-0021a	ST12_RIYVXINN_52Y-0021a	ST13_RIYVXINN_52Y-0021a	ST14_RIYVXINN_52Y-0021a	ST21_ISLOCA
Release Category		1.1	1.2	1.3	1.4	2.1
OALARM (uncovery) (hrs)	Core Uncovery (IEVNT(49))	1.67	1.67	1.67	1.67	8.35E-02
OALARM (uncovery) (s)	Core Uncovery (IEVNT(49))	6012	6012	6012	6012	301
PLHEAT (watts)		3.87E+07	1.45E+07	3.87E+07	1.45E+07	8.91E+06
PLHITE (meters)	TDPLHITE	18.44	18.44	18.44	18.44	2.13
RELFRC	FREL(1)	1.00E+00	3.70E-01	1.00E+00	3.70E-01	1.00E+00
	FREL(2)	4.04E-02	7.05E-02	4.42E-02	5.95E-02	3.44E-01
	FREL(3)	1.06E-02	3.05E-02	1.52E-02	2.88E-02	3.18E-01
	FREL(4)	2.16E-04	3.93E-05	2.46E-04	8.30E-05	2.64E-02
	FREL(5)	3.94E-03	3.36E-03	3.94E-03	3.41E-03	1.03E-02
	FREL(6)	1.51E-02	2.90E-02	2.39E-02	2.40E-02	3.20E-01
	FREL(7)	1.11E-03	4.81E-04	1.12E-03	5.14E-04	2.27E-02
	FREL(8)	6.05E-06	2.43E-06	1.17E-05	1.40E-05	3.53E-03
	FREL(9)	4.11E-05	1.10E-05	1.40E-04	1.65E-04	3.85E-02
	FREL(10)	3.58E-01	1.16E-02	4.19E-01	8.85E-03	2.56E-01
	FREL(11)	7.85E-11	4.97E-06	1.92E-06	2.26E-05	2.96E-03
	FREL(12)	3.60E-15	1.14E-08	1.35E-06	1.50E-06	3.25E-04
PDELAY (hrs)		73.80	2.25	73.80	2.25	0.5
PDELAY(s)		265680	8100	265680	8100	1800
PLUDUR (hrs)		74.18	51.39	74.17	29.95	36.07
PLUDUR (s)		267048	185004	267012	107820	129852
End of Release (hrs)		147.98	53.64	147.97	32.20	36.57

Table E.3-13: MAAP Output for MACCS2 (continued)

Davis-Besse MAAP Case ID		ST22_ISLOCA	ST31_AXI1A_4	ST32_AXI1A_4	ST33_AXI1A_4	ST34_AXI1A_4
Release Category		2.2	3.1	3.2	3.3	3.4
OALARM (uncovery) (hrs)	Core Uncovery (IEVNT(49))	8.37E-02	8.34E-02	8.39E-02	8.34E-02	8.39E-02
OALARM (uncovery) (s)	Core Uncovery (IEVNT(49))	301	300	302	300	302
PLHEAT (watts)		6.68E+06	2.22E+06	2.61E+06	2.22E+06	2.61E+06
PLHITE (meters)	TDPLHITE	2.13	45.42	45.42	45.42	45.42
RELFRC	FREL(1)	9.40E-01	1.00E+00	9.66E-01	9.90E-01	9.92E-01
	FREL(2)	2.89E-01	6.34E-03	3.03E-01	4.55E-03	3.11E-01
	FREL(3)	2.65E-01	3.61E-03	2.52E-01	3.42E-03	2.73E-01
	FREL(4)	4.51E-03	1.17E-04	3.58E-03	1.17E-04	1.56E-02
	FREL(5)	1.25E-02	2.10E-04	1.23E-02	2.09E-04	1.31E-02
	FREL(6)	2.75E-01	5.94E-03	2.56E-01	4.36E-03	2.81E-01
	FREL(7)	1.01E-02	2.40E-04	9.08E-03	2.39E-04	1.48E-02
	FREL(8)	1.64E-04	2.55E-06	1.36E-04	2.70E-06	2.82E-03
	FREL(9)	6.61E-04	1.20E-05	6.05E-04	1.35E-05	3.24E-02
	FREL(10)	1.55E-01	8.62E-03	2.02E-01	4.85E-03	2.62E-01
	FREL(11)	2.12E-05	1.99E-07	0.00E+00	1.97E-07	2.28E-03
	FREL(12)	1.48E-07	3.34E-10	7.26E-08	1.76E-08	2.80E-04
PDELAY (hrs)		0.58	0.42	0.42	0.42	0.42
PDELAY(s)		2088	1512	1512	1512	1512
PLUDUR (hrs)		14.62	49.52	29.58	9.08	43.38
PLUDUR (s)		52632	178272	106488	32688	156168
End of Release (hrs)		15.20	49.94	30	9.5	43.8

Table E.3-13: MAAP Output for MACCS2 (continued)

Davis-Besse MAAP Case ID		ST41_AXI1A_4	ST42_AXI1A_4	ST43_AXI1A_4	ST44_AXI1A_4	ST51_SIYYFYYN _36Y-002
Release Category		4.1	4.2	4.3	4.4	5.1
OALARM (uncovery) (hrs)	Core Uncovery (IEVNT(49))	8.38E-02	8.38E-02	8.38E-02	8.38E-02	6.68E-01
OALARM (uncovery) (s)	Core Uncovery (IEVNT(49))	302	302	302	302	2405
PLHEAT (watts)		9.17E+05	2.24E+05	6.77E+05	2.10E+05	3.17E+06
PLHITE (meters)	TDPLHITE	2.13	2.13	2.13	2.13	45.42
RELFRC	FREL(1)	5.31E-01	5.60E-01	4.69E-01	5.49E-01	9.99E-01
	FREL(2)	9.90E-05	1.96E-02	7.73E-05	6.16E-03	8.28E-03
	FREL(3)	2.94E-06	1.08E-02	2.64E-06	3.35E-03	1.26E-03
	FREL(4)	2.95E-14	7.14E-05	4.92E-09	2.08E-03	3.27E-07
	FREL(5)	2.89E-13	1.98E-04	7.90E-09	1.02E-04	7.66E-07
	FREL(6)	6.33E-05	1.30E-02	7.46E-05	4.06E-03	1.50E-03
	FREL(7)	7.53E-14	1.81E-04	3.43E-08	9.72E-04	1.03E-06
	FREL(8)	3.75E-16	2.71E-06	6.18E-10	3.69E-04	1.03E-08
	FREL(9)	9.19E-16	1.05E-05	7.73E-09	4.38E-03	2.25E-08
	FREL(10)	7.54E-04	7.62E-03	6.81E-04	1.44E-02	8.29E-04
	FREL(11)	0.00E+00	0.00E+00	1.40E-04	7.60E-04	5.33E-08
	FREL(12)	0.00E+00	0.00E+00	6.44E-09	3.25E-05	1.57E-11
PDELAY (hrs)		12.5	0.58	14	0.58	4.1
PDELAY(s)		45000	2088	50400	2088	14760
PLUDUR (hrs)		37.21	49.14	35.71	49.16	22.50
PLUDUR (s)		133956	176904	128556	176976	81000
End of Release (hrs)		49.71	49.72	49.71	49.74	26.6

Table E.3-13: MAAP Output for MACCS2 (continued)

Davis-Besse MAAP Case ID		ST52_TINYNINN_53Y	ST53_SIYYFYYN_36Y-002	ST54_TINYNINN_53Y	ST61_TINYNINN_53Y	ST62_TINYNINN_53Y
Release Category		5.2	5.3	5.4	6.1	6.2
OALARM (uncovery) (hrs)	Core Uncovery (IEVNT(49))	9.17E-01	6.68E-01	9.17E-01	9.17E-01	9.17E-01
OALARM (uncovery) (s)	Core Uncovery (IEVNT(49))	3301	2405	3301	3301	3301
PLHEAT (watts)		1.09E+07	2.83E+06	9.59E+06	7.35E+07	1.14E+08
PLHITE (meters)	TDPLHITE	45.42	45.42	45.42	2.13	2.13
RELFRC	FREL(1)	9.65E-01	9.85E-01	9.95E-01	9.93E-01	9.91E-01
	FREL(2)	1.16E-01	7.31E-03	1.11E-01	3.12E-02	7.16E-02
	FREL(3)	1.72E-01	1.03E-03	1.71E-01	1.55E-02	3.15E-02
	FREL(4)	1.71E-04	1.02E-06	2.50E-02	3.09E-05	3.22E-05
	FREL(5)	1.09E-03	7.15E-07	9.33E-04	1.88E-04	1.13E-04
	FREL(6)	9.05E-02	1.33E-03	9.12E-02	1.62E-02	2.50E-02
	FREL(7)	1.19E-03	1.24E-06	1.21E-02	1.58E-04	2.55E-04
	FREL(8)	1.71E-05	2.26E-07	4.10E-03	1.84E-06	3.14E-06
	FREL(9)	9.08E-05	2.79E-06	6.37E-02	1.50E-05	1.53E-05
	FREL(10)	2.48E-02	7.82E-04	1.91E-01	5.90E-03	1.48E-02
	FREL(11)	0.00E+00	1.97E-07	6.29E-03	1.92E-08	4.41E-07
	FREL(12)	0.00E+00	2.68E-08	3.68E-04	0.00E+00	0.00E+00
PDELAY (hrs)		2.08	4.1	2.08	2.34	2.42
PDELAY(s)		7488	14760	7488	8424	8712
PLUDUR (hrs)		12.92	5.90	48.02	1.46	37.58
PLUDUR (s)		46512	21240	172872	5256	135288
End of Release (hrs)		15	10	50.1	3.8	40

Table E.3-13: MAAP Output for MACCS2 (continued)

Davis-Besse MAAP Case ID		ST63_TINYNINN_53Y	ST64_TINYNINN_53Y	ST71_AXI1A_4	ST72_AXI1A_4	ST73_TINYNINN_53Y
Release Category		6.3	6.4	7.1	7.2	7.3
OALARM (uncovery) (hrs)	Core Uncovery (IEVNT(49))	9.17E-01	9.17E-01	8.35E-02	8.35E-02	3.51
OALARM (uncovery) (s)	Core Uncovery (IEVNT(49))	3301	3301	301	301	12636
PLHEAT (watts)		6.10E+07	1.16E+08	3.02E+07	2.79E+07	2.82E+07
PLHITE (meters)	TDPLHITE	2.13	2.13	45.42	45.42	45.42
RELFRC	FREL(1)	9.99E-01	9.95E-01	1.00E+00	1.00E+00	9.99E-01
	FREL(2)	2.03E-03	1.81E-02	2.34E-02	9.93E-02	2.35E-04
	FREL(3)	1.82E-04	1.93E-03	3.96E-03	9.82E-03	1.55E-05
	FREL(4)	1.12E-09	2.30E-04	1.14E-08	1.80E-08	4.68E-10
	FREL(5)	2.72E-09	1.73E-06	1.98E-08	6.96E-08	6.48E-10
	FREL(6)	9.44E-04	2.82E-03	2.31E-02	4.08E-02	4.91E-05
	FREL(7)	8.60E-09	1.07E-04	2.90E-08	5.37E-08	2.23E-09
	FREL(8)	2.99E-10	4.29E-05	3.40E-10	4.35E-10	2.34E-11
	FREL(9)	8.30E-10	8.20E-04	1.29E-09	1.59E-09	2.07E-10
	FREL(10)	3.91E-04	2.92E-02	1.50E-02	1.55E-02	1.07E-05
	FREL(11)	2.36E-07	1.57E-04	0.00E+00	0.00E+00	0.00E+00
	FREL(12)	8.30E-11	6.50E-06	0.00E+00	0.00E+00	0.00E+00
PDELAY (hrs)		11.9	11	29.0	33.3	35.6
PDELAY(s)		42840	39600	104400	119880	128160
PLUDUR (hrs)		48.07	48.04	47.95	47.98	48.00
PLUDUR (s)		173052	172944	172620	172728	172800
End of Release (hrs)		59.97	59.04	76.95	81.28	83.6

Table E.3-13: MAAP Output for MACCS2 (continued)

Davis-Besse MAAP Case ID		ST74_TINYNINN_53Y	ST75_AXI1A_4	ST76_AXI1A_4	ST77_TINYNINN_53Y	ST78_TINYNINN_53Y
Release Category		7.4	7.5	7.6	7.7	7.8
OALARM (uncovery) (hrs)	Core Uncovery (IEVNT(49))	3.51	8.35E-02	8.36E-02	3.51	3.51
OALARM (uncovery) (s)	Core Uncovery (IEVNT(49))	12636	301	301	12636	12636
PLHEAT (watts)		2.80E+07	2.01E+07	2.36E+07	1.93E+07	2.45E+07
PLHITE (meters)	TDPLHITE	45.42	45.42	45.42	45.42	45.42
RELFRC	FREL(1)	1.00E+00	9.97E-01	9.93E-01	9.99E-01	9.89E-01
	FREL(2)	3.36E-04	2.06E-03	2.16E-02	1.07E-05	1.76E-02
	FREL(3)	3.72E-05	1.80E-05	3.78E-03	5.97E-07	1.69E-03
	FREL(4)	4.69E-10	1.16E-08	6.37E-06	5.66E-10	1.10E-05
	FREL(5)	6.48E-10	2.07E-08	1.05E-06	6.96E-10	2.95E-07
	FREL(6)	2.01E-05	8.58E-04	7.49E-03	1.41E-06	1.37E-03
	FREL(7)	2.23E-09	2.95E-08	4.95E-06	2.35E-09	5.78E-06
	FREL(8)	2.34E-11	3.44E-10	8.77E-07	4.32E-11	1.86E-06
	FREL(9)	2.07E-10	1.33E-09	1.15E-05	4.86E-10	3.64E-05
	FREL(10)	1.24E-06	5.66E-03	2.08E-02	2.05E-06	1.90E-02
	FREL(11)	0.00E+00	8.59E-08	1.96E-03	2.74E-08	1.03E-03
	FREL(12)	0.00E+00	7.84E-12	2.22E-07	7.57E-12	2.96E-07
PDELAY (hrs)		40.8	42	37.2	50.9	42.5
PDELAY(s)		146880	151200	133920	183240	153000
PLUDUR (hrs)		11.90	48.01	48.08	48.01	48.04
PLUDUR (s)		42840	172836	173088	172836	172944
End of Release (hrs)		52.7	90.01	85.28	98.91	90.54

Table E.3-13: MAAP Output for MACCS2 (continued)

Davis-Besse MAAP Case ID		ST81_AXI1a_4	ST82_AXI1a_4	ST91_AXI1A_4	ST92_AXI1A_4
Release Category		8.1	8.2	9.1	9.2
OALARM (uncovery) (hrs)	Core Uncovery (IEVNT(49))	8.36E-02	8.36E-02	8.36E-02	8.34E-02
OALARM (uncovery) (s)	Core Uncovery (IEVNT(49))	301	301	301	300
PLHEAT (watts)		8.71E+06	9.78E+07	2.63E+02	3.30E+02
PLHITE (meters)	TDPLHITE	0.00	0.00	45.42	45.42
RELFRC	FREL(1)	9.32E-01	9.93E-01	1.47E-03	1.51E-03
	FREL(2)	5.57E-06	4.84E-02	5.66E-07	4.51E-05
	FREL(3)	5.03E-07	9.35E-03	4.64E-07	3.38E-05
	FREL(4)	2.14E-08	7.64E-05	2.09E-08	6.01E-07
	FREL(5)	1.85E-07	7.72E-06	2.00E-07	1.78E-06
	FREL(6)	3.17E-06	2.85E-02	5.26E-07	3.49E-05
	FREL(7)	5.58E-08	4.63E-05	5.74E-08	1.36E-06
	FREL(8)	5.32E-10	1.14E-05	5.11E-10	1.99E-08
	FREL(9)	1.60E-09	1.53E-04	1.60E-09	9.08E-08
	FREL(10)	2.50E-05	1.92E-02	5.22E-07	3.09E-05
	FREL(11)	9.20E-09	1.86E-03	0.00E+00	3.29E-09
	FREL(12)	3.00E-12	2.22E-06	0.00E+00	8.69E-12
PDELAY (hrs)		33.2	15.4	0.33	0.5
PDELAY(s)		119520	55440	1188	1800
PLUDUR (hrs)		47.90	47.95	15.27	49.42
PLUDUR (s)		172440	172620	54972	177912
End of Release (hrs)		81.1	63.35	15.6	49.92

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.3-18	**E-96**	**Farm & Nonfarm (last 2) columns**

ER Table E.3-18, "Economic Data," the 'Farmland Property Value for the Region' and 'Nonfarm Property Value for the Region' columns, are replaced in their entirety, to read as follows:

Table E.3-18: Economic Data

Region Name, State	Fraction of Land Devoted to Farming in Region	Fraction of Farm Sales Resulting from Dairy in Region	Total Annual Farm Sales for the Region ($/hectare)	Farmland Property Value for the Region ($/hectare)	Nonfarm Property Value for the Region ($/person)
Crawford, OH	0.854	0.044	1301	7,907	34,979
Erie, OH	0.522	0.025	1186	9,869	82,281
Fulton, OH	0.709	0.086	1802	8,859	59,090
Hancock, OH	0.729	0.032	1007	8,033	56,893
Huron, OH	0.697	0.055	1507	8,540	42,523
Lorain, OH	0.395	0.106	2612	11,120	71,245
Lucas, OH	0.289	0.000	1881	10,751	68,848
Ottawa, OH	0.706	0.019	990	7,144	117,709
Sandusky, OH	0.694	0.024	1081	7,630	54,067
Seneca, OH	0.764	0.021	985	7,719	37,526
Wood, OH	0.698	0.044	1125	8,300	70,940
Lenawee, MI	0.727	0.244	1142	7,902	23,140
Monroe, MI	0.591	0.011	1547	9,454	34,958
Wayne, MI	0.045	0.000	4074	19,128	28,338

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.3-19	**E-96**	**Entire table**

ER Table E.3-19, "MACCS2 Economic Parameters Used in CHRONC," is replaced in its entirety, to read as follows:

Table E.3-19: MACCS2 Economic Parameters Used in CHRONC

Variable	Description	Value (in Davis-Besse model)
DPRATE	Property depreciation rate (/year)	0.20
DSRATE	Investment rate of return (/year)	0.12
POPCST	Population relocation cost ($/person)	$9,750/person
CDFRM0	Cost of farm decontamination for various levels of decontamination ($/hectare)	$1,096.90/hectare, $2,437.50/hectare
CDNFRM	Cost of non-farm decontamination per person for various levels of decontamination ($/person)	$5,850/person, $15,600/person
DLBCST	Average cost of decontamination labor ($/person-year)	$68,250/person-year

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.3-20	**E-97**	**Release Category rows 2.1 and 2.2**

ER Table E.3-20, "Frequency Vector," 'Release Category rows 2.1 and 2.2, 'Frequency (/year)' column data are reversed, and the rows are revised to read as follows:

Release Category	Frequency (/year)	Percent
2.1	~~*5.4E-08*~~ *6.0E-09*	0.06%
2.2	~~*6.0E-09*~~ *5.4E-08*	0.53%

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.3-21	**E-98**	**Entire table**

ER Table E.3-21, "Base Case Results for Internal Events at 50 Miles," previously revised by FENOC letter dated June 24, 2011 (ML11180A233), is replaced in its entirety, and now reads:

Table E.3-21: Base Case Results for Internal Events at 50 Miles

Release Category	Whole Body Dose (50, rem)/yr	Economic Impact (50, $)/yr
1.1	4.29E-02	5.70E+01
1.2	2.41E-02	5.15E+01
1.3	1.28E+00	2.23E+03
1.4	2.12E-03	4.20E+00
2.1	4.89E-02	6.78E+01
2.2	3.05E-01	5.10E+02
3.1	2.43E-03	1.44E+00
3.2	1.52E-04	2.62E-01
3.3	1.90E-05	9.05E-03
3.4	1.27E-02	1.84E+01
4.1	2.47E-05	5.25E-03
4.2	4.69E-02	5.92E+01
4.3	3.03E-07	6.00E-05
4.4	1.05E-02	1.54E+01
5.1	9.69E-03	4.26E+00
5.2	1.15E-02	2.46E+01
5.3	7.67E-04	3.78E-01
5.4	6.51E-03	7.70E+00
6.1	5.68E-04	8.62E-01
6.2	5.68E-05	1.13E-01
6.3	9.45E-04	3.09E-01
6.4	2.44E-02	1.36E+01
7.1	2.17E-05	4.55E-02
7.2	1.05E-03	2.65E+00
7.3	3.83E-08	5.32E-06
7.4	2.24E-05	3.98E-03
7.5	5.72E-06	2.05E-03
7.6	2.11E-02	1.55E+01
7.7	4.25E-08	8.86E-07
7.8	3.68E-02	1.60E+01
8.1	1.32E-04	2.33E-03

Release Category	Whole Body Dose (50, rem)/yr	Economic Impact (50, $)/yr
8.2	2.08E-01	4.88E+02
9.1	1.92E-03	2.03E-06
9.2	1.75E-02	2.25E+00
Total	**2.12E+00**	**3.59E+03**

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.3-22	**E-99**	**Entire table**

ER Table E.3-22, "Base Case Consequence Input to SAMA Analysis," previously revised by FENOC letter dated June 24, 2011 (ML11180A233), is replaced in its entirety, and now reads:

Table E.3-22: Base Case Consequence Input to SAMA Analysis

Release Category	Whole Body Dose (50, rem)	Economic Impact (50, $)
1.1	1.95E+06	2.59E+09
1.2	1.85E+06	3.96E+09
1.3	2.17E+06	3.78E+09
1.4	1.77E+06	3.50E+09
2.1	8.15E+06	1.13E+10
2.2	5.64E+06	9.45E+09
3.1	9.73E+05	5.77E+08
3.2	5.44E+06	9.36E+09
3.3	7.58E+05	3.62E+08
3.4	7.48E+06	1.08E+10
4.1	2.47E+04	5.25E+06
4.2	1.38E+06	1.74E+09
4.3	2.75E+04	5.45E+06
4.4	1.36E+06	2.00E+09
5.1	3.34E+05	1.47E+08
5.2	3.02E+06	6.47E+09
5.3	2.74E+05	1.35E+08
5.4	7.31E+06	8.65E+09
6.1	1.29E+06	1.96E+09
6.2	1.72E+06	3.41E+09
6.3	2.10E+05	6.87E+07
6.4	7.86E+05	4.39E+08
7.1	1.55E+06	3.25E+09
7.2	1.85E+06	4.65E+09
7.3	1.74E+04	2.42E+06
7.4	9.32E+03	1.66E+06
7.5	2.12E+05	7.60E+07
7.6	1.11E+06	8.17E+08
7.7	1.18E+03	2.46E+04
7.8	3.76E+05	1.63E+08
8.1	2.10E+03	3.70E+04

Release Category	Whole Body Dose (50, rem)	Economic Impact (50, $)
8.2	1.60E+06	3.75E+09
9.1	2.53E+02	2.67E-01
9.2	1.25E+04	1.61E+06
Total	**6.07E+07**	**9.34E+10**

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Tables E.3-23 through E.3-32	**E-100 & E-101**	**Entire Tables (10 tables)**

ER Tables E.3-23 through E.3-32, "Comparison of Base Case and Case [XX]," previously revised by FENOC letter dated June 24, 2011 (ML11180A233), are replaced in their entirety, with the exception of Table E.3-27, "Comparison of Base Case and Case M2," which is no longer used and is deleted, and the tables read as follows:

Table E.3-23: Comparison of Base Case and Case S1

	Internal Events		
	Base	**S1**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	2.12E+00	2.32E+00	9.4%
Economic Impact (50) ($/yr)	3.59E+03	3.92E+03	9.2%

Table E.3-24: Comparison of Base Case and Case S2

	Internal Events		
	Base	**S2**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	2.12E+00	1.88E+00	-11.3%
Economic Impact (50) ($/yr)	3.59E+03	3.20E+03	-10.9%

Table E.3-25: Comparison of Base Case and Case S3

	Internal Events		
	Base	**S3**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	2.12E+00	2.18E+00	2.8%
Economic Impact (50) ($/yr)	3.59E+03	3.59E+03	0.0%

Table E.3-26: Comparison of Base Case and Case M1

	Internal Events		
	Base	**M1**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	2.12E+00	2.11E+00	-0.5%
Economic Impact (50) ($/yr)	3.59E+03	3.63E+03	1.1%

Table E.3-27: ~~*Comparison of Base Case and Case M2*~~
[Table E.3-27 is not used, and is deleted.]

Table E.3-28: Comparison of Base Case and Case A1

	Internal Events		
	Base	**A1**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	2.12E+00	2.05E+00	-3.3%
Economic Impact (50) ($/yr)	3.59E+03	3.40E+03	-5.3%

Table E.3-29: Comparison of Base Case and Case A2

	Internal Events		
	Base	**A2**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	2.12E+00	2.12E+00	0.0%
Economic Impact (50) ($/yr)	3.59E+03	3.59E+03	0.0%

Table E.3-30: Comparison of Base Case and Case A3

	Internal Events		
	Base	**A3**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	2.12E+00	2.12E+00	0.0%
Economic Impact (50) ($/yr)	3.59E+03	3.59E+03	0.0%

Table E.3-31: Comparison of Base Case and Case E1

	Internal Events		
	Base	**E1**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	2.12E+00	2.11E+00	-0.5%
Economic Impact (50) ($/yr)	3.59E+03	3.59E+03	0.0%

Table E.3-32: Comparison of Base Case and Case E2

	Internal Events		
	Base	**E2**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	2.12E+00	1.62E+00	-23.6%
Economic Impact (50) ($/yr)	3.59E+03	2.16E+03	-39.8%

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.3-33	**E-101**	**Entire table**

Based on the revised response to RAI 4.d, ER Table E.3-33, "Comparison of Base Case and Case E3," previously added in response to RAI 6.j (see revised response to RAI 6.j in this letter) by FENOC letter dated June 24, 2011 (ML11180A233), reads as follows:

Table E.3-33: Comparison of Base Case and Case E3

	Internal Events		
	Base	**S1**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	2.12E+00	2.12E+00	0.0%
Economic Impact (50) ($/yr)	3.59E+03	3.59E+03	0.0%

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.4-1	**E-101**	**Entire table**

Based on the responses to RAIs 3.c and 4.b (see FENOC letter dated June 24, 2011 (ML11180A233), and the revised SAMA Analysis, ER Table E.4-1, "Total Cost of Severe Accident Impact," is revised to read as follows:

Table E.4-1: Total Cost of Severe Accident Impact

APE	*$52,025*
AOC	*$44,049*
AOE	$4,340
AOSC	$266,279
Severe Accident Impact (Internal Events)	*$366,693*
Fire, Seismic, Other	*$1,686,788*
Maximum Benefit (Internal Events, Fire, Seismic, Other)	*$2,053,481*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.5-4	**E-144 thru 154**	**6 rows revised; 1 new row**

In response to RAIs 5.c and 5.f (see FENOC letter dated June 24, 2011 (ML11180A233), ER Table E.5-4, "List of Initial SAMA Candidates," is revised as follows:

Table E.5-4: List of Initial SAMA Candidates

SAMA Candidate Identifier	SAMA Candidate Description	Derived Benefit	Source
CB-21	Install pressure measurements between the two DHR suction valves in the line from the RCS hot leg.	This SAMA candidate would provide indication of failure of inboard isolation valves allowing time to initiate mitigating actions to prevent ISLOCA.	[2, Table 14] *[Table E.5-2]*
CC-19	Provide automatic switchover of HPI and LPI suction from the BWST to containment sump for LOCAs.	This SAMA candidate will increase the reliability of switchover of suction from the BWST to the containment sump by providing both manual and automatic switchover.	*[Table E.5-1]*
CP-19	Install a redundant containment fan system.	This SAMA candidate would increase containment heat removal ability. *SAMA candidate CP-19 was added as a variation to CP-18 to provide a redundant containment cooling function, in the form of containment fan coolers.*	*Davis-Besse containment cooling design*
CW-24	Replace the standby CCW pump with a pump diverse from the other two CCW pumps.	This SAMA candidate would improve CCW reliability by reducing the likelihood of a CCF of all three CCW pumps.	*[Table E.5-1]* *[Table E.5-2]*
CW-25	Provide the ability to cool make-up pumps using fire water in the event of loss of CCW.	This SAMA candidate would allow continued injection of RCP seal water in the event of loss of CCW.	*[Table E.5-1]* *[Table E.5-2]*
FW-16	Perform surveillances on manual valves used for backup AFW pump suction.	This SAMA candidate would improve the success probability for providing an alternate water supply to the AFW pumps.	~~*[2, Table 14]*~~ *[Table E.5-1]* *[Table E.5-2]*
OT-09R	*Provide operator training with PRA-identified high risk important human actions to be emphasized in training.*	*PRA results show that operator actions are significant contributors to overall plant risk. By highlighting those operator actions shown to have the highest risk importance, the reliability of those actions will be improved.*	*Table E.5-2*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.6-1	**E-155 thru 180**	**8 rows revised, and 1 new row**

In response to RAIs 5.c, 5.g, 5.h, 6.b (for RAIs 5.h and 6.b, see FENOC letter dated June 24, 2011 (ML11180A233), and 6.k, and to align with current FENOC plans and with the discussions in the ER regarding the steam generator replacement schedule, Table E.6-1, "Qualitative Screening of SAMA Candidates," is revised as follows:

Table E.6-1: Qualitative Screening of SAMA Candidates

SAMA ID	Modification (Potential Enhancement)	Screening Criteria	Basis for Screening/Modification Enhancements
CB-08	Improve operator training on ISLOCA coping.	~~*Criterion E*~~ ~~*Subsumed*~~ *Criterion B* *Already Implemented*	~~*This SAMA would reduce the risk of ISLOCA events by improving the likelihood of timely identification and diagnosis of ISLOCA events and thereby increasing the likelihood of successful mitigating actions. This SAMA will be subsumed in CB-07.*~~ *Davis-Besse has several procedures in place to address small and interfacing system LOCAs. Operators receive training on LOCAs, and there are a number of indications to support the likelihood and timely identification and diagnosis of ISLOCA events (including tank level indications, lifting relief valves, and running sump pumps).*
CB-09	Institute a maintenance practice to perform a 100% inspection of steam generator tubes during each refueling outage.	Criterion D Very Low Benefit	*Davis-Besse is scheduled to replace the steam generators in ~~2013~~ 2014, which would result in inspecting new steam generator tubes.* Therefore, this SAMA candidate is considered very low benefit for Davis-Besse.
CB-10	Replace steam generators with a new design.	Criterion B Already Implemented	*Davis-Besse is scheduled to replace the steam generators in ~~2013~~ 2014.* Therefore, the intent of the SAMA candidate has already been implemented at Davis-Besse.

Table E.6-1: Qualitative Screening of SAMA Candidates (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CC-08	Add the ability to automatically align ECCS to recirculation mode upon BWST depletion.	Criterion E Subsumed	Davis-Besse currently has the ability to initiate automatic switchover from the BWST to the containment sump on low BWST level, but this feature has been deactivated. ~~*The cost would by minor to reactivate this feature.*~~ This SAMA candidate will be subsumed in SAMA candidate CC-19.
CC-19	Provide automatic switchover of HPI and LPI suction from the BWST to containment sump for LOCAs.	Criterion F Considered for Further Evaluation	Davis-Besse currently has the ability to initiate automatic switchover from the BWST to the containment sump on low BWST level, but this feature has been deactivated. ~~*The cost would by minor to reactivate this feature.*~~ Therefore, this SAMA candidate is considered for further evaluation.
FL-01	Improve inspection of rubber expansion joints on main condenser.	~~*Criterion D*~~ ~~*Very Low Benefit*~~ *Criterion B* *Already Implemented*	~~*Based on the top 100 cutsets and component basic event importance, circulating water breaks are not a significant risk contributor at Davis-Besse.*~~ *The circulating water joints are currently inspected during outages, and include both interior and exterior inspections. Exterior inspections of the visible portion of the expansion joint are performed during Engineering system walkdowns and Operator tours. Additionally, the expansion joints are periodically replaced.*
OT-05	Increase training and operating experience feedback to improve operator response.	~~*Criterion D*~~ ~~*Very Low Benefit*~~ *Criterion B* *Already Implemented*	~~*No deficiencies in operator training or feedback are identified.*~~ *FENOC provides PRA information, such as risk-significant initiating events, high worth operator actions and high worth equipment, to various departments, including Operations Training, and presents this information on posters throughout the plant.*
OT-07	Install secondary side guard pipes up to the MSIVs.	Criterion D Very Low Benefit	*Steam line breaks are not a significant contributor to CDF or LERF.* The derived benefit would not justify the implementation cost required.
OT-09R	*Provide operator training with PRA-identified high risk important human actions to be emphasized in training.*	*Criterion B* *Already Implemented*	*Davis-Besse provides PRA information such as risk significant initiating events, high worth operator actions and high worth equipment. This information is provided to various departments and is presented on posters throughout the plant.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.7-2	**E-183 – 185**	**Entire table**
ER Table E.7-3	**E-186**	**Entire table**
ER Table E.7-5	**E-188**	**Entire table**
ER Table E.8-1	**E-189 - 190**	**Entire table**

Based on the responses to RAIs 3.c and 4.b (see FENOC letter dated June 24, 2011 (ML11180A233), and the revised SAMA Analysis, ER Tables E.7-2, E.7-3, E.7-5 and E.8-1 are replaced in their entirety, to read as shown on the following pages:

Table E.7-2: Internal Events Benefit Results for Analysis Case

Case	Maximum Benefit	AC/DC-01 (DCBattery)	AC/DC-03 (Battery Charger)	AC/DC-14 (GasTurbineGen)
Off-site Annual Dose (rem)	2.12E+00	2.08E+00	1.87E+00	1.78E+00
Off-site Annual Property Loss ($)	3.59E+03	3.53E+03	3.17E+03	3.02E+03
Comparison CDF	----	1.0E-05	1.0E-05	1.0E-05
Comparison Dose (rem)	----	2.12E+00	2.12E+00	2.12E+00
Comparison Cost ($)	----	3.59E+03	3.59E+03	3.59E+03
Enhanced CDF	----	9.4E-06	7.8E-06	9.0E-06
Reduction in CDF	----	**6.00%**	**22.00%**	**10.00%**
Reduction in Off-site Dose	----	**1.89%**	**11.79%**	**16.04%**
Immediate Dose Savings (On-site)	$810	$49	$178	$81
Long Term Dose Savings (On-site)	$3,530	$212	$777	$353
Total Accident Related Occupational Exposure **(AOE)**	$4,340	$260	$955	$434
Cleanup/Decontamination Savings (On-site)	$132,362	$7,942	$29,120	$13,236
Replacement Power Savings (On-site)	$133,917	$8,035	$29,462	$13,392
Averted Costs of On-site Property Damage **(AOSC)**	$266,279	$15,977	$58,581	$26,628
Total On-site Benefit	**$270,619**	**$16,237**	**$59,536**	**$27,062**
Averted Public Exposure **(APE)**	$52,025	$982	$6,135	$8,344
Averted Off-site Damage Savings **(AOC)**	$44,049	$736	$5,153	$6,994
Total Off-site Benefit	**$96,074**	**$1,718**	**$11,288**	**$15,338**
Total Benefit (On-site + Off-site)	**$366,693**	**$17,955**	**$70,824**	**$42,399**

Table E.7-2: Internal Events Benefit Results for Analysis Case (continued)

Case	AC/DC-19 (FireWaterBackup)	AC/DC-21 (RepairBreakers)	AC/DC-25 (DedDCPower)	AC/DC-26 (Generator TDAFW)
Off-site Annual Dose (rem)	2.08E+00	2.11E+00	2.05E+00	2.05E+00
Off-site Annual Property Loss ($)	3.54E+03	3.58E+03	3.48E+03	3.48E+03
Comparison CDF	1.0E-05	1.0E-05	1.0E-05	1.0E-05
Comparison Dose (rem)	2.12E+00	2.12E+00	2.12E+00	2.12E+00
Comparison Cost ($)	3.59E+03	3.59E+03	3.59E+03	3.59E+03
Enhanced CDF	9.8E-06	9.7E-06	8.5E-06	8.5E-06
Reduction in CDF	**2.00%**	**3.00%**	**15.00%**	**15.00%**
Reduction in Off-site Dose	**1.89%**	**0.47%**	**3.30%**	**3.30%**
Immediate Dose Savings (On-site)	$16	$24	$121	$121
Long Term Dose Savings (On-site)	$71	$106	$529	$529
Total Accident Related Occupational Exposure **(AOE)**	$87	$130	$651	$651
Cleanup/Decontamination Savings (On-site)	$2,647	$3,971	$19,854	$19,854
Replacement Power Savings (On-site)	$2,678	$4,018	$20,088	$20,088
Averted Costs of On-site Property Damage **(AOSC)**	$5,326	$7,988	$39,942	$39,942
Total On-site Benefit	**$5,412**	**$8,119**	**$40,593**	**$40,593**
Averted Public Exposure **(APE)**	$982	$245	$1,718	$1,718
Averted Off-site Damage Savings **(AOC)**	$614	$123	$1,350	$1,350
Total Off-site Benefit	**$1,595**	**$368**	**$3,068**	**$3,068**
Total Benefit (On-site + Off-site)	**$7,007**	**$8,487**	**$43,660**	**$43,660**

Table E.7-2: Internal Events Benefit Results for Analysis Case (continued)

Case	AC/DC-27 (SBO_DieselTank)	CB-21 (DHR_valves)	CC-01 (HPI_System)	CC-04 (LPI_pump)
Off-site Annual Dose (rem)	2.12E+00	2.00E+00	2.10E+00	2.12E+00
Off-site Annual Property Loss ($)	3.59E+03	3.40E+03	3.58E+03	3.59E+03
Comparison CDF	1.0E-05	1.0E-05	1.0E-05	1.0E-05
Comparison Dose (rem)	2.12E+00	2.12E+00	2.12E+00	2.12E+00
Comparison Cost ($)	3.59E+03	3.59E+03	3.59E+03	3.59E+03
Enhanced CDF	1.0E-05	1.0E-05	1.0E-05	1.0E-05
Reduction in CDF	**0.00%**	**0.00%**	**0.00%**	**0.00%**
Reduction in Off-site Dose	**0.00%**	**5.66%**	**0.94%**	**0.00%**
Immediate Dose Savings (On-site)	$0	$0	$0	$0
Long Term Dose Savings (On-site)	$0	$0	$0	$0
Total Accident Related Occupational Exposure **(AOE)**	$0	$0	$0	$0
Cleanup/Decontamination Savings (On-site)	$0	$0	$0	$0
Replacement Power Savings (On-site)	$0	$0	$0	$0
Averted Costs of On-site Property Damage **(AOSC)**	$0	$0	$0	$0
Total On-site Benefit	**$0**	**$0**	**$0**	**$0**
Averted Public Exposure **(APE)**	$0	$2,945	$491	$0
Averted Off-site Damage Savings **(AOC)**	$0	$2,331	$123	$0
Total Off-site Benefit	**$0**	**$5,276**	**$614**	**$0**
Total Benefit (On-site + Off-site)	**$0**	**$5,276**	**$614**	**$0**

Table E.7-2: Internal Events Benefit Results for Analysis Case (continued)

Case	CC-05 (LPI_Diesel_pump)	CC-19 (BWST_to_Sump)	HV-01 (Redundant_HVAC)	HV-03 (Backup_fans)
Off-site Annual Dose (rem)	2.12E+00	2.12E+00	2.11E+00	2.11E+00
Off-site Annual Property Loss ($)	3.59E+03	3.59E+03	3.59E+03	3.59E+03
Comparison CDF	1.0E-05	1.0E-05	1.0E-05	1.0E-05
Comparison Dose (rem)	2.12E+00	2.12E+00	2.12E+00	2.12E+00
Comparison Cost ($)	3.59E+03	3.59E+03	3.59E+03	3.59E+03
Enhanced CDF	1.0E-05	9.9E-06	1.0E-05	1.0E-05
Reduction in CDF	**0.00%**	**1.00%**	**0.00%**	**0.00%**
Reduction in Off-site Dose	**0.00%**	**0.00%**	**0.47%**	**0.47%**
Immediate Dose Savings (On-site)	$0	$8	$0	$0
Long Term Dose Savings (On-site)	$0	$35	$0	$0
Total Accident Related Occupational Exposure **(AOE)**	$0	$43	$0	$0
Cleanup/Decontamination Savings (On-site)	$0	$1,324	$0	$0
Replacement Power Savings (On-site)	$0	$1,339	$0	$0
Averted Costs of On-site Property Damage **(AOSC)**	$0	$2,663	$0	$0
Total On-site Benefit	**$0**	**$2,706**	**$0**	**$0**
Averted Public Exposure **(APE)**	$0	$0	$245	$245
Averted Off-site Damage Savings **(AOC)**	$0	$0	$0	$0
Total Off-site Benefit	**$0**	**$0**	**$245**	**$245**
Total Benefit (On-site + Off-site)	**$0**	**$2,706**	**$245**	**$245**

Table E.7-3: Total Benefit Results for Analysis Case

	Maximum Benefit	AC/DC-01 (DCBattery)	AC/DC-03 (Battery Charger)	AC/DC-14 (GasTurbineGen)	AC/DC-19 (FireWaterBackup)	AC/DC-21 (RepairBreakers)
Internal Events	$366,693	$17,955	$70,824	$42,399	$7,007	$8,487
Fires, Seismic, Other	$1,686,788	$82,593	$325,792	$195,037	$32,234	$39,039
Total Benefit	$2,053,481	$100,547	$396,617	$237,436	$39,242	$47,525

	AC/DC-25 (DedDCPower)	AC/DC-26 (Generator_TDAFW)	AC/DC-27 (SBO_DieselTank)	CB-21 (DHR_valves)	CC-01 (HPI_System)
Internal Events	$43,660	$43,660	$0	$5,276	$614
Fires, Seismic, Other	$200,837	$200,837	$0	$24,270	$2,822
Total Benefit	$244,497	$244,497	$0	$29,546	$3,436

	CC-04 (LPI_pump)	CC-05 (LPI_Dieselpump)	CC-19 (BWST_to_Sump)	HV-01 (Redundant_HVAC)	HV-03 (Backup_fans)
Internal Events	$0	$0	$2,706	$245	$245
Fires, Seismic, Other	$0	$0	$12,448	$1,129	$1,129
Total Benefit	$0	$0	$15,155	$1,374	$1,374

Table E.7-5: Final Results of Cost Benefit Evaluation

SAMA Candidate ID	Modification	Estimated Benefit	2009 Estimate Cost	Conclusion
AC/DC-01	Provide additional DC battery capacity.	$100,547	$1,750,000	Not Cost Effective
AC/DC-03	Add a portable, diesel-driven battery charger to existing DC system.	$396,617	$330,000	Cost Effective
AC/DC-14	Install a gas turbine generator.	$237,436	$2,000,000	Not Cost Effective
AC/DC-19	Use fire water system as a backup source for diesel cooling.	$39,242	$700,000	Not Cost Effective
AC/DC-21	Develop procedures to repair or replace failed 4kV breakers.	$47,525	$100,000	Not Cost Effective
AC/DC-25	Provide a dedicated DC power system (battery/battery charger) for the TDAFW control valve and NNI-X for steam generator level indication.	$244,497	$2,000,000	Not Cost Effective
AC/DC-26	Provide an alternator/generator that would be driven by each TDAFW pump.	$244,497	$2,000,000	Not Cost Effective
AC/DC-27	Increase the size of the SBO fuel oil tank.	$0	$550,000	Not Cost Effective
CB-21	Install pressure measurements between the two DHR suction valves in the line from the RCS hot leg.	$29,546	$550,000	Not Cost Effective
CC-01	Install an independent active or passive HPI system.	$3,436	$6,500,000	Not Cost Effective
CC-04	Add a diverse LPI system.	$0	$5,500,000	Not Cost Effective
CC-05	Provide capability for alternate LPI via diesel-driven fire pump.	$0	$6,500,000	Not Cost Effective
CC-19	Provide automatic switchover of HPI and LPI suction from the BWST to containment sump for LOCAs.	$15,155	$1,500,000	Not Cost Effective
HV-01	Provide a redundant train or means of ventilation.	$1,374	$50,000	Not Cost Effective
HV-03	Stage backup fans in switchgear rooms.	$1,374	$400,000	Not Cost Effective

Table E.8-1: Final Results of the Sensitivity Cases

SAMA Candidate ID	Repair Case	Low Discount Rate Case	High Discount Rate Case	On-site Dose Case	On-site Cleanup Case	2009 Estimated Cost	Conclusion
AC/DC-01	$64,551	$152,033	$69,662	$102,023	$115,372	$1,750,000	Not Cost Effective
AC/DC-03	$264,628	**$600,596**	$276,817	**$402,026**	**$450,974**	$330,000	**Cost Effective**
AC/DC-14	$177,442	$361,238	$169,575	$239,895	$262,144	$2,000,000	Not Cost Effective
AC/DC-19	$27,243	$59,518	$27,604	$39,734	$44,183	$700,000	Not Cost Effective
AC/DC-21	$29,527	$71,774	$32,727	$48,263	$54,938	$100,000	Not Cost Effective
AC/DC-25	$154,505	$369,476	$168,897	$248,186	$281,559	$2,000,000	Not Cost Effective
AC/DC-26	$154,505	$369,476	$168,897	$248,186	$281,559	$2,000,000	Not Cost Effective
AC/DC-27	$0	$0	$0	$0	$0	$550,000	Not Cost Effective
CB-21	$29,546	$45,615	$22,616	$29,546	$29,546	$550,000	Not Cost Effective
CC-01	$3,436	$5,304	$2,630	$3,436	$3,436	$6,500,000	Not Cost Effective
CC-04	$0	$0	$0	$0	$0	$5,500,000	Not Cost Effective
CC-05	$0	$0	$0	$0	$0	$6,500,000	Not Cost Effective
CC-19	$9,155	$22,864	$10,383	$15,401	$17,625	$1,500,000	Not Cost Effective
HV-01	$1,374	$2,122	$1,052	$1,374	$1,374	$50,000	Not Cost Effective
HV-03	$1,374	$2,122	$1,052	$1,374	$1,374	$400,000	Not Cost Effective

Table E.8-1: Final Results of the Sensitivity Cases (continued)

SAMA Candidate ID	Replacement Power Case	Multiplier Case	Evacuation Speed	95th Percentile CDF	2009 Estimated Cost	Conclusion
AC/DC-01	$130,750	$143,639	$100,547	$145,794	$1,750,000	Not Cost Effective
AC/DC-03	**$507,358**	**$566,596**	**$396,617**	**$575,095**	$330,000	**Cost Effective**
AC/DC-14	$287,773	$339,195	$237,436	$344,283	$2,000,000	Not Cost Effective
AC/DC-19	$49,309	$56,060	$39,242	$56,901	$700,000	Not Cost Effective
AC/DC-21	$62,626	$67,893	$47,525	$68,912	$100,000	Not Cost Effective
AC/DC-25	$320,003	$349,282	$244,497	$354,521	$2,000,000	Not Cost Effective
AC/DC-26	$320,003	$349,282	$244,497	$354,521	$2,000,000	Not Cost Effective
AC/DC-27	$0	$0	$0	$0	$550,000	Not Cost Effective
CB-21	$29,546	$42,209	$29,546	$42,842	$550,000	Not Cost Effective
CC-01	$3,436	$4,908	$3,436	$4,982	$6,500,000	Not Cost Effective
CC-04	$0	$0	$0	$0	$5,500,000	Not Cost Effective
CC-05	$0	$0	$0	$0	$6,500,000	Not Cost Effective
CC-19	$20,188	$21,649	$15,155	$21,974	$1,500,000	Not Cost Effective
HV-01	$1,374	$1,963	$1,374	$1,993	$50,000	Not Cost Effective
HV-03	$1,374	$1,963	$1,374	$1,993	$400,000	Not Cost Effective

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.11	**E-194**	**New references**

In response to RAI 3.c, ER Section E.11, "References," is revised to include two new references cited in revised ER Section E.4.5, as follows:

39. Nuclear Regulatory Commission, "Requests for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit 1, License Renewal Application," Accession Number ML110910566, April 20, 2011.

40. Nuclear Regulatory Commission, Results of Safety/Risk Assessment of Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," Accession Number ML100270582, September 7, 2010.

ATTACHMENT 6

Letter from K. Byrd, FirstEnergy, to NRC Document Control Desk, "Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4613), Environmental Report Severe Accident Mitigation Alternatives Analysis, and License Renewal Application Amendment No. 10 (June 24, 2011)

FENOC
FirstEnergy Nuclear Operating Company

Davis-Besse Nuclear Power Station
5501 N. State Route 2
Oak Harbor, Ohio 43449

June 24, 2011
L-11-154

10 CFR 54

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Davis-Besse Nuclear Power Station, Unit No. 1
Docket No. 50-346, License Number NPF-3
Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4613)
Environmental Report Severe Accident Mitigation Alternatives Analysis, and
License Renewal Application Amendment No. 10

By letter dated August 27, 2010, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102450565), FirstEnergy Nuclear Operating Company (FENOC) submitted an application pursuant to Title 10 of the *Code of Federal Regulations*, Part 54 for renewal of Operating License NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS). By letter dated April 20, 2011 (ADAMS Accession No. ML110910566), the Nuclear Regulatory Commission (NRC) requested additional information to complete its review of the License Renewal Application (LRA).

The Attachment provides the FENOC reply to the NRC request for additional information. The NRC request is shown in bold text followed by the FENOC response. The Enclosure provides Amendment No. 10 to the DBNPS LRA. The due date for this reply was changed from June 20 to June 24, 2011, as mutually agreed to by Ms. Paula Cooper, NRC Environmental Project Manager, on June 17, 2011.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Clifford I. Custer, Fleet License Renewal Project Manager, at 724-682-7139.

A145
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 24, 2011.

Sincerely,

Kendall W. Byrd
Director, Site Performance Improvement

Attachment:
Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application, Environmental Report, Attachment E, Severe Accident Mitigation Alternatives Analysis

Enclosure:
Amendment No. 10 to the DBNPS License Renewal Application

cc: NRC DLR Project Manager
NRC DLR Environmental Project Manager
NRC Region III Administrator

cc: w/o Attachment or Enclosure
NRC DLR Director
NRR DORL Project Manager
NRC Resident Inspector
Utility Radiological Safety Board

Item 1

Provide the following information regarding the Level 1 Probabilistic Risk Assessment (PRA) used for the Severe Accident Mitigation Alternatives (SAMA) analysis:

Question RAI 1.a

Environmental Report (ER) Section E.3.1.1.2 explains that the SAMA evaluation is based on an updated version of the Davis-Besse Revision 4 PRA model that takes advantage of a 2008 "gap self assessment." This model, referred to as the "SAMA Analysis Model" represents a "freeze date" of July 9, 2009 for plant configuration, August 1, 2006 for component failure data and initiating event data, April 30, 2007 for equipment availability, and January 1, 2006 for non-Maintenance Rule unavailability. Identify any changes to the plant (physical and procedural modifications) since July 9, 2009 that could have a significant impact on the results of the PRA and/or SAMA analyses. Provide an assessment of their impact on the PRA and on the results of the SAMA evaluation.

RESPONSE RAI 1.a

As discussed in the response to RAI 1.c, below, plant changes are tracked for subsequent PRA updates. While there have been some plant changes since the SAMA model, no changes have been identified that have a significant impact on the PRA results or SAMA evaluation. Based on FirstEnergy Nuclear Operating Company (FENOC) Nuclear Operating Business Practice NOBP-CC-6001, "PRA Model Management," plant changes are evaluated to determine if they would cause a change of greater than 10 percent core damage frequency (CDF), or greater than 20 percent large early release frequency (LERF); there have been no changes that meet this criteria since the SAMA model.

Question RAI 1.b

ER Section E.3.1.1.2 describes the PRA model history from 1993, when the IPE was issued, to July 2009 when the SAMA Analysis Model became effective. This section specifically discusses the model updates to Revision 2, 3, 4, and the SAMA Analysis Model. This section does not discuss the model revision from the IPE to the Revision 0, when the largest decrease in internal events CDF occurred (i.e., a decrease from 6.6E-05/yr to 1.4E-05/yr), or the update to Revision 1. Also, the reason for the drop in internal events CDF between the Revision 3 and 4 PRA models of approximately a factor of three is not apparent from the model update discussion. Provide a discussion of the PRA model changes that most impacted the change in total internal events CDF for the Revision 0, 1, and 4 PRA models. Also provide the effective dates of the Revision 0, 1, and 2 PRA models.

RESPONSE RAI 1.b

The second underlined section in Environmental Report (ER) Section E.3.1.1.2 is titled "Davis-Besse PRA, Revision 0 – CDF = 1.4E-05/yr to Revision 2 CDF = 1.7E-05/yr and LERF = 7.3E-08/yr"; this section discusses changes made in the PRA Revision 0, PRA Revision 1 and PRA Revision 2 models, collectively. The largest decrease in risk, from the Individual Plant Examination (IPE) CDF of 6.5E-05/yr, to the PRA Revision 0 CDF of 1.4E-05/yr, is primarily due to a reduction in transient frequencies for the reactor/turbine trip (T_1) and the loss of main feedwater (T_2) transients. The slight increase in risk from the PRA Revision 0 CDF of 1.4E-05/yr, to the PRA Revision 1 CDF of 1.6E-05/yr is primarily associated with a data update.

Subsequent PRA revisions are also discussed in ER Section E.3.1.1.2. The decrease in risk from the PRA Revision 3 CDF of 1.3E-05/yr, to the PRA Revision 4 CDF of 4.7E-06/yr is primarily associated with increasing the time operators have to trip the reactor coolant pumps (RCPs) following a loss of seal cooling (supplied by the Component Cooling Water (CCW) System), and a data update.

The IPE was completed in February 1993; the PRA Quantification Notebook was signed off in March 1999 for PRA Revision 0, August 1999 for PRA Revision 1, October 1999 for PRA Revision 2, and September 2007 for PRA Revision 4. These are the effective dates for each PRA revision.

Question RAI 1.c

Provide a brief description of the quality control process used for controlling changes to the PRA, including the process of monitoring potential plant changes, tracking items that may lead to model changes, making model changes (including frequency for model updates), documenting changes, software quality control, independent reviews, and qualification of PRA staff.

RESPONSE RAI 1.c

PRA quality control is covered under: 1) FENOC Nuclear Operating Program Manual NOPM-CC-6000, "Probabilistic Risk Assessment Program;" and 2) FENOC Nuclear Operating Business Practice NOBP-CC-6001, "Probabilistic Risk Assessment Model Management." Both procedures identify requirements for maintaining and updating the PRA models and applications and both were developed in accordance with Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, to assure the PRA is technically acceptable and supports risk-informed applications in accordance with NRC regulatory guidelines. Specific elements of NOPM-CC-6000 include:

- Requirement 4.2.1, that the PRA be maintained and updated to represent the as-designed, as-built, as-operated plant.

- Requirement 4.2.4, that the PRA be conducted by qualified personnel with industry recognized levels of capabilities and skills in PRA, commensurate with EPRI TR-1011981, "Development of PRA Qualification and Curriculum," dated September 2005. In addition, Section 5 of NOPM-CC-6000 addresses Qualifications and Training. This section requires that PRA team members meet the PRA Analyst qualification requirements of Job Performance Requirement (JPR) 2.4; this JPR addresses the requirements for a Davis-Besse Analyst, requring completion of the EPRI PRA Fundamentals course (or equivalent), required reading, as well as mentor discussions and proficiency demonstrations. One pre-requisite for JPR 2.4 is completion of the Davis-Besse Engineering Support Personnel orientation training, and the Davis-Besse systems training.

- Section 6.2 on Self-Assessments; they are to be performed on as as-needed basis, and at an interval not to exceed 3 years. The results of Self-Assessments and issues identified are evaluated and changes incorporated into the PRA Program as appropriate as required by the FENOC Self-Assessment/Benchmarking procedure.

- Section 7.3 on PRA Software and Computer Control. All PRA software and computers shall be under configuration control as specified in the PRA Software and Computer Control Plan in accordance with NOP-SS-1001, FENOC Administrative Program for Computer Related Activities; this provides

requirements for verification of all approved versions of PRA specific software and computers.

- Section 8.4 on PRA Software QA Requirements. All PRA software shall comply with NOP-SS-1001, FENOC Administrative Program for Computer Related Activities.

- Section 9.1 on PRA Program Records that identifies specific PRA documentation that should be maintained.

Specific elements of NOBP-CC-6001 include:

- Section 5.1.1 on Tracking and Disposition of Plant Changes. Each site is required to have a system for identifying, tracking and dispositioning plant changes that may affect the PRA model; at Davis-Besse, this is done in accordance with NOP-CC-2004, "Design Interface (DIE) Reviews and Evaluations," in which proposed plant changes are routed to the PRA group to identify if the change will impact the PRA. The DIE forms are contained in the Configuration Management Interface System (CMIS). Similarly, NOP-SS-3001, "Procedure Review and Approval," requires a cross-disciplinary review of proposed procedure changes.

- Section 5.1.2 on Reference Model Updates. This section identifies those items that should be reviewed for possible PRA updates, including plant changes, data, and industry experience.

- Section 5.3 on PRA Revisions; PRA models are expected to be revised every other refueling cycle.

- Section 5.4 on Models and Documentation.

Question RAI 1.d

ER Section E.3.1.1.2 identifies a Babcock and Wilcox (B&W) owner's group peer review of the internal events Level 1 and LERF PRA models performed on November 8, 1999 and states that no Level A and 18 Level B supporting requirements findings were identified. The ER further explains that following the review a Revision 3 PRA was issued to "close gaps to the draft industry standards." It is not clear from this statement whether all Level B findings were resolved by the Revision 3 PRA model. Section E.3.3 of the ER also discusses a B&W owner's group peer review that was finished in March 2000 which states that there were no Level A findings, and presents 5 Level B findings, three of which are closed and two that are still open. It is not clear whether this is the same B&W

owner's group peer review comments described in Section 3.1.1.2, and if it is, why there are discrepancies in the two descriptions. The ER also states that in 2008 a "gap self assessment" was performed using a team of industry peers and internal staff that identified four Level A findings and 23 Level B findings associated with not meeting Capability Category 2 requirements of the 2005 ASME PRA standard. It is not clear from the description what the scope of this "gap self assessment" included. The ER does not identify any other peer reviews, technical reviews, or self assessments of the PRA. In light of these issues, provide the following:

i. **Clarify whether there were one or two B&W owner's group peer reviews performed in late 1999 and early 2000 and the differences (e.g., scope) between these reviews if there were two. Clarify whether any Level A or B findings remain unresolved from this peer review (or these peer reviews) and if so, provide an assessment of their impact on the SAMA evaluation.**

ii. **Clarify the scope of the 2008 "gap self assessment" including whether it covered Level 1 and 2 internal events, internal flooding, and the high winds hazard. Also, identify the open Level A and B findings from this self assessment and provide an assessment of their impact on the SAMA evaluation.**

iii. **Provide a summary of the scope of any other PRA model internal and external reviews, a discussion of each unresolved finding, and an assessment of the impact of all unresolved findings on the SAMA evaluation.**

RESPONSE RAI 1.d

1.d.i

There was one B&W peer review performed; it was performed in late 1999, and the report was issued in early 2000. There were no Level A findings, and of the 18 Level B level findings, 13 were closed prior to implementation of the Mitigating Systems Performance Index (MSPI) Basis Document; 4 were closed in the SAMA model; and the 1 remaining finding recommended additional sensitivity studies be performed.

As noted in ER Section E.3.3, FENOC plans to include sensitivity studies in Revision 5 of the PRA. The sensitivity studies recommended in EPRI Report 1016737, Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments, address Human Error Probabilities (HEP) and Common Cause Factors (CCF). Since the basic event importance results for the Level 1 PRA and LERF (discussed in ER Sections E.5.4 and E.5.5, as well as E.3.1.1.1 and E.3.2.1) include Human Failure Events (HFEs), components, and initiating events, and these items were reviewed and

considered in identifying SAMAs, no new or additional insights are expected that would have a significant impact on the SAMA evaluation.

1.d.ii

The scope of the 2008 gap self-assessment included the following PRA technical areas: initiating events; accident sequences evaluation; success criteria; systems analysis; human reliability analysis; data analysis; quantification; and, maintenance and update.

As discussed in ER Section E.3.1.1.2, the 2008 gap self-assessment was targeted at identifying 'gaps' to meet Capability Category II (of the PRA standard ASME RA-Sb-2005). Also, as discussed in ER E.3.1.1.2, the Davis-Besse SAMA model has all level A and B findings addressed.

1.d.iii

Other than those reviews described in paragraphs i and ii above, the PRA team is not aware of any other peer reviews of the PRA model.

Question RAI 1.e

ER Section E.3.1.1.1 states that the Davis-Besse Level 1 PRA internal events CDF is estimated to be 9.2E-6/yr, but further explains that if high winds and internal flooding is included that the CDF is estimated to be 9.8E-6/yr. Regarding the internal events CDF, provide the following:

i. **The ER provides a caveat about the "tornado high winds" analysis in Section E.3.1.2.3 saying that the model does not include tornado-generated missiles. Based on the top 100 cutsets presented in Table E.5-1, the contribution to the total CDF from tornadoes does not appear to be significant (i.e. Cutset #1 = 3.0E-8/yr, #30 = 2.8E-8/r, #69 =1.2E-8/yr, and #87 = 1.2E-8/yr). The NRC staff notes that the contribution to the internal events CDF from internal flooding is typically included in the internal events CDF whereas the contribution from high winds is generally not included. In light of this and given the high winds analysis is not complete, provide the internal events CDF including flooding but excluding high winds.**

ii. **ER Table E.3-1 presents dominant internal event sequences by initiating event and their percentage contribution to CDF that includes a contribution**

from internal flooding (i.e., F3AM and F7L). The calculated contribution percentages in Table E.3-1 appear to be based on a CDF of 9.2E-06/yr. This is consistent with the CDF reported in Section E.3.1.1.1 for the internal events CDF that does not include internal flooding and external wind, rather than the CDF of 9.2E-06/yr that does includes internal flooding and external winds. Clarify this apparent discrepancy. Also, clarify which model the Level 2 PRA was based on (i.e., with or without inclusion of internal flooding and external wind).

RESPONSE RAI 1.e

1.e.i

ER Section E.3.1.1.1, second paragraph is revised to read:

The Davis-Besse Level 1 PRA internal event CDF (including internal flooding) is 9.2E-6/yr, and, when also including high winds, the CDF is 9.8E-6/yr.

1.e.ii

As discussed above, the Davis-Besse Level 1 PRA internal event CDF, including internal flooding, is 9.2E-6/yr. The Davis-Besse Level 2 PRA is based on the Level 1 internal event PRA, including internal flooding and tornados/high winds, with a CDF of 9.8E-6/yr.

See the Enclosure to this letter for the revision to the DBNPS LRA.

Question RAI 1.f

In ER Table E.3-1, initiating event T2B-1 listed as "SP6A fails to throttle" and T2A-1 listed as "SP6B fails to throttle" appear to have mismatching nomenclature and descriptions. Also it is not clear which valves are being referred to or what their function is in the plant. Initiating event T2A-2 listed as "FICICS35B fails high" and T2B-2 listed as "FICICS35A fails high" also appear to have mismatching nomenclature and descriptions. It is also unclear for these initiating events which components are being referred to or what their function is in the plant. Clarify these apparent discrepancies and provide layman descriptions for these four initiators.

RESPONSE RAI 1.f

The nomenclature is based on plant numbering guidelines. Davis-Besse typically assigns train 1 valves "B" suffixes, and train 2 valves "A" suffixes. Valves SP6A and SP6B are the main feedwater flow control valves: FICICS35A and FICICS35B are the associated flow controllers for the valves. Events T2A-1 and T2A-2 represent main feedwater overfeeds on steam generator 1: T2A-1 is associated with valve SP6B and T2A-2 is associated with its flow controller FICICS35B. Events T2B-1 and T2B-2 represent main feedwater overfeeds on steam generator 2: T2B-1 is associated with valve SP6A and T2B-2 is associated with its flow controller FICICS35A.

Item 2

Provide the following information relative to the Level 2 analysis:

Question RAI 2.a

ER Section E.3.1.1.1 states that the Level 1 PRA quantification was performed using a "truncation cutoff" of 5E-13/yr, but no reference is made to the Level 2 truncation cutoff. Provide the Level 2 PRA truncation cutoff.

RESPONSE RAI 2.a

The Level 2 PRA was also performed at a truncation of 5E-13/yr. ER Section E.3.2.1 is revised to include this truncation value.

See the Enclosure to this letter for the revision to the DBNPS LRA.

Question RAI 2.b

ER section E.3.2.1 states that "The CET provides the framework for evaluating containment failure modes and conditions that would affect the magnitude of the release." The ER also explains that "The probabilities of the CET end states were quantified for each PDS." However, the Containment Event Tree (CET) is not presented in the ER nor is a description of its structure and composition provided. Provide the CET or a description of the CET used in the Level 2 analysis. Include in the response a discussion of how the CET top events were selected and how branch points probabilities were determined, including how phenomenological versus system failure mode branch point probabilities were determined.

RESPONSE RAI 2.b

The Containment Event Tree (CET) provides the framework for evaluating containment failure modes and conditions that would affect the magnitude of a release. The Davis-Besse CET was developed from a Babcock & Wilcox Owners Group (B&WOG) generic CET and refined to address phenomena that could have a significant impact on RCS integrity, containment response and eventual release from containment. Table 2.b-1, below, identifies the top events and branches in the CET.

Table 2.b-1: Containment Event Tree Events and Branches

CET Events	Branches
A: Arrest of Core Damage In-Vessel	Success – core cooling restored in time to prevent vessel failure or steam generator tube creep rupture Failure – cooling not restored
R: Submerged-Vessel Cooling of Core Debris	Success – reactor cavity flooding prevents vessel failure Failure – vessel breach
V: Ctmt Bypass	No Bypass Bypassed – ISLOCA or SGTR (i.e., direct radionuclide release)
B1: Ctmt Isolated	Containment Isolated Isolation failure
B2: Isolation Failure	Small – containment did not depressurize appreciably Large – containment depressurizes
E: Early Ctmt Failure Prevented	No Early Failure Early Failure – no potential for fission product scrubbing
C: Ex-Vessel Cooling	Debris Cooled – prevents core-concrete interaction Debris Uncooled – basemat or sidewall failure
D: Ctmt Sidewall	No Sidewall Failure Sidewall Failure
L: Late Ctmt Failure	No Late Failure Late Failure
F: Late Revaporization Release	No Revaporization Revaporization
S: Fission Product Scrubbing	Scrubbed Unscrubbed

Branch probabilities in the CET were determined based on a consideration of phenomena and elements of the associated core damage bin and plant damage state. Phenomena probabilities were estimated based on references (e.g., NUREG-1150), sensitivity studies, and judgment. House events were used to determine applicable CET branches based on the core damage bin and plant damage state.

Question RAI 2.c

ER Section 3.1.1.2 states that an explicit LERF model was added to the PRA. ER Section 3.2.1 states that 14 additional PDSs were added to better define the status of certain containment systems. Clarify how the Level 2 model used in the SAMA evaluation differs from the IPE analysis.

RESPONSE RAI 2.c

ER Section E.3.2.2 discusses the Level 2 PRA model changes since the IPE. One of the most significant changes is the level of detail reflected in the plant-damage states (PDS), and the manner in which their frequencies were calculated. Nearly 500 PDS were defined to accommodate the core-damage bins and the various combinations of system states that could affect subsequent Containment response. In the SAMA Level 2 PRA, 14 additional PDS were added to better define the status of Containment systems to support CET quantification. Since the IPE, a framework was also established to allow all of the PDS frequencies to be calculated in a manner that could be readily repeated for sensitivity studies and applications.

Another change involved developing a probability distribution for Containment failure as a function of internal pressure. The analysis investigated various mechanisms for Containment failure to identify those that might limit its capacity. The expected yield strength was calculated and a distribution was developed based on variability in the materials used, and uncertainties. A second distribution was developed to apply to scenarios in which pressurization would occur over a long period of time, such that the heating of the Containment might reduce the strength of the Containment shell.

Reviews were also made of new analytical studies completed since the IPE. One review identified a change in the treatment of the potential for a rupture of a steam generator tube to be induced due to the transport of hot gases to the steam generators during meltdown of the core (e.g., PDS TIN_18Y).

Other changes include enhancements in quantification capabilities, and changes in the Level 1 PRA, including: updates based on plant changes, procedure changes, and maintenance changes; system enhancements to support applications such as the Maintenance Rule; updates to the SGTR analysis based on emergency operating procedure (EOP) changes; updates in initiating event frequencies and component failure rates based on plant experience; and improvements in technical methods such as the Human Reliability Analysis.

The LERF quantification process has also been simplified; the process allows LERF cutsets to be generated without the lengthy quantification process required to a complete the Level 2 analysis.

Question RAI 2.d

Identify the version of MAAP used in the SAMA analysis.

RESPONSE RAI 2.d

MAAP 4.0.6 was used in the SAMA analysis.

Question RAI 2.e

Identify the release categories that compose the large early release frequency (LERF) from those presented in Table E.3-4 (Release Categories 1.1 through 9.2). Confirm that the identified release categories are those reviewed in Table E.5-3 (Basic Event LERF Importance).

RESPONSE RAI 2.e

ER Table E.3-4 identifies the Release Categories and descriptions; LERF was calculated using the following Release Categories: 1.2 and 1.4 (steam generator tube rupture (SGTR)), 2.1 and 2.2 (interfacing system loss of coolant accident (ISLOCA)), 3.2 and 3.4 (Large Isolation containment failure), 5.2 and 5.4 (Early containment failure), and 6.1 and 6.2 (Sidewall containment failure).

A re-review of LERF importance and ER Table E.5-3, "Basic Event LERF Importance" (pg E-136), based on these Release Categories, identified a few discrepancies: the omission of two events (UHAMUHPE and FMFWTRIP); and the inclusion of two extra events (ZHABWMUE and NORCVRT3, which are just below the risk reduction worth (RRW) cutoff). There are also some slight discrepancies in the rankings, Fussell-Vesely (F-V) importance measures, and RRW importance measures (e.g., in the ER, QHAMDFPE has a F-V of 5.96E-02 and a RRW of 1.063, but should have a F-V of 6.80E-02 and RRW of 1.073, and should be immediately preceding FLC0100F and not immediately following FLC0100F). In addition, FVV011AT should be defined as 'AVV fails to reseat after steam release' (and not fails to reseat after SGTR).

ER Table E.5-3 is revised to correct the identified discrepancies.

See the Enclosure to this letter for the revision to the DBNPS LRA.

Item 3

Provide the following information with regard to the treatment and inclusion of external events in the SAMA analysis:

Question RAI 3.a

For each of the four dominant fire areas identified in ER Section E.3.1.2.1, provide the following:

i. **Explain what measures have already been taken to reduce risk. Include in the response specific consideration of improvements to detection systems, enhancements to suppression capabilities, changes that would improve cable separation and drain separation, and monitoring and controlling the quantity of combustible materials in critical process areas.**

ii. **Review to identify potential SAMA candidates to reduce fire risk. Provide a Phase I and II assessment, as applicable, of each SAMA candidate. If no SAMA candidates are identified, explain why the fire CDF cannot be further reduced in a cost effective manner through implementation of SAMAs specific to fire events.**

RESPONSE RAI 3.a

3.a.i

A large portion of fire risk is associated with control of combustibles, both transient and permanent; this is primarily accomplished through proper management of maintenance of fire detection and suppression systems, and configuration control of the fire design features, such as fire barriers. Following the issuance of the Individual Plant Examination for External Events (IPEEE), Davis-Besse began utilizing a software tool, the Fire Risk Management Program, that tracks inoperable or degraded fire protection features as well as manages transient combustible loads and travel paths. This software is maintained by the site Fire Marshall and controlled by operations procedures: DB-FP-0007, "Control of Transient Combustibles", DB-FP-0018, "Control of Ignition Sources", and DB-FP-0009,"Fire Protection Impairment and Fire Watch".

The Fire Risk Management Program is a software tool designed to capture fire protection requirements along with expert knowledge to provide real time fire risk assessment and management. This tool allows users at all levels to understand fire risks and ensure the application of appropriate risk management techniques, and includes establishing fire watches, limiting hot work and prohibiting transient combustibles.

3.a.ii

The four dominant areas identified in ER Section E.3.1.2.1 are Q.01, S.01, X.01, and FF.01. The dominant contributors to risk in three of these areas are the motor-driven feedwater pump (MDFP), the Auxiliary Feedwater (AFW) System, and the pilot-operated relief valve (PORV). The fourth area, the Control Room, area FF.01, is further divided into "control room not evacuated" and "control room evacuated". In both cases, the dominant contributor is a loss of feedwater, and AFW, MDFP, and the PORV are again the main contributors to risk. When the control room is evacuated, the ability to feed and bleed is greatly hindered, so the importance of the PORV is diminished for control room evacuation scenarios.

A review of SAMAs was performed with the intent of identifying modifications that could improve fire-related risk. As described above, the fire risk is generally driven by loss of all feedwater and inability to perform feed and bleed; the fire initiator feeds into the transient event tree and core damage sequences are governed by a loss of feedwater or inability to perform feed and bleed cooling. The following SAMAs apply and the alternatives and evaluations are bounded by the existing analysis; these SAMAs were evaluated as 'Already Implemented' in ER Table E.6-1:

- CC-16
- FW-02
- FW-08
- FW-09
- FW-10
- FW-11

No additional SAMAs were identified unique to fire risk.

Question RAI 3.b

ER Section E.3.1.2.1 presents the four fire areas identified in the IPEEE that had an estimated CDF above the screening criteria of 1E-06/yr. It also presents the summation of those fire area CDFs to be 2.5E-05/yr which is then used as the basis to develop an external events multiplier. The IPEEE SER (Enclosure 3, Section 2.1.7) explains that the total frequency of the fire area CDFs which had been screened out after detailed analysis (some of which had revised CDFs greater than 1E-06/yr) is 3.8E-06/yr, which results in a total fire CDF of 2.9E-05/yr. Identify the fire compartments that were screened after detailed analysis and the

corresponding CDFs and provide a review of these fire compartments for potential SAMAs.

RESPONSE RAI 3.b

The fire compartments that were screened are delineated in Table 4.2.3.2 of the IPEEE. There are fifteen compartments that start with A.07 and end with Y.02. One column in this table describes the fire effects. The effects are identical to those described in response to RAI 3.a.ii, above; they are associated with secondary side actions including a loss of feedwater and actions pertaining to the AFW System. The SAMAs associated with these actions have been evaluated in response to RAI 3.a.ii; no new SAMAs were identified unique to these compartments or fire risk.

Question RAI 3.c

ER Section E.3.1.2.4 presents the basis for an external events multiplier of 3 based on a "conservatively" estimated fire CDF of 2.5E-05/yr developed using the FIVE methodology and the assumption that a "realistic" fire CDF is a factor of 3 less than this FIVE-produced fire CDF. The NRC staff disagrees that a fire CDF produced using the FIVE screening methodology is necessarily conservative in light of more recent research and guidance on hot short probabilities (i.e., NUREG/CR-6850). The NRC staff particularly notes that the minimal or non-treatment of hot shorts in the IPEEE FIVE analysis may more than offset other conservatisms in the FIVE analysis. Based on this, and the previous RAI, the NRC staff believes the best estimate of the fire CDF for Davis-Besse is 2.9E-05/yr. In addition, the USGS issued updated seismic hazard curves for much of the U.S. in 2008. Using this data, the NRC staff estimated a "weakest link model" seismic CDF for Davis-Besse of 6.7E-06/yr (see NRC Information Notice 2010-18 regarding Generic Issue 199). Based on a fire CDF of 2.9E-05/yr, a seismic CDF of 6.7E-06/yr, and an internal events CDF of 9.8E-06/yr, the NRC staff estimates the external events multiplier to be 3.6. In light of this, provide a revised SAMA evaluation using an external events multiplier of 3.6 or alternatively provide justification for an evaluation of a different multiplier based on this updated USGS information.

RESPONSE RAI 3.c

Based on the information provided in the RAI, an updated external events multiplier was calculated for Davis-Besse. The updated external events multiplier includes risk contribution from fire, seismic, and other hazard groups. The risk contribution for the fire and seismic hazard groups was determined by a ratio between the hazard group

CDF and the internal events CDF as shown in the equations below. The risk contribution from the other hazard group was conservatively assumed to be equivalent to the internal events contribution. Therefore, the other hazard group multiplier is 1.0.

Fire Hazard Multiplier:

$$\frac{\text{Fire CDF}}{\text{Internal Events CDF}} = \frac{2.9\text{x}10^{-5}/\text{yr}}{1.0\text{x}10^{-5}/\text{yr}} = 2.90$$

Seismic Hazard Multiplier:

$$\frac{\text{Seismic CDF}}{\text{Internal Events CDF}} = \frac{6.7\text{x}10^{-6}/\text{yr}}{1.0\text{x}10^{-5}/\text{yr}} = 0.67$$

To determine the multiplier to account for fire, seismic, and other hazard groups, the three individual multipliers were summed, resulting in a multiplier of 4.6. The cost-benefit evaluation was updated using an external event multiplier of 4.6. The updated maximum benefit for Davis-Besse is $1,955,223. Based on the updated maximum benefit, one SAMA candidate, AC/DC-03 (add a portable diesel-driven battery charger to the direct current (DC) system) was determined to be cost-beneficial.

ER Section E.3.1.2.4, "External Event Severe Accident Risk," is deleted based on the response to this RAI. ER Section E.4.5, "Total Cost of Severe Accident Risk," is revised to explain the updated external events multiplier. ER Tables E.4-1, E.7-2, E.7-3, E.7-5, and E.8-1 are revised to reflect the revised cost-benefit results.

See the Enclosure to this letter for the revision to the DBNPS LRA.

Item 4

Provide the following information concerning the Level 3 analysis:

Question RAI 4.a

Regarding ER Section E.3.4.7, clarify that the core inventory is based on the rated thermal power of 2,817 MWt and, if not, provide justification for the thermal power used.

RESPONSE RAI 4.a

The core inventory source term analysis used to generate Environmental Report Table E.3-17, "Davis-Besse Core Inventory (Full Core at EOC; 177FAs)," incorporates a two percent uncertainty in core power, or:

P=1.02 x 2772 megawatts thermal (Mwt) = 2827.44 Mwt

Question RAI 4.b

Table 2.6-1 identifies that the year 2000 population living within the 50-mile site boundary is 2,375,624. Table E.3-11 identifies that the escalated population to year 2040 is only 2,227,192. The year 2040 population was stated to be a 4.7% escalation per decade from year 2000. Clarify this discrepancy. Also, in ER Section E.3.4.2, the statement that actual population within the 50-mile radius decreases appears to be incorrect. This statement appears to apply only to the US population groups within a 20-mile radius. Clarify that this understanding is correct.

RESPONSE RAI 4.b

The discrepancy in the 2000 population within a 50-mile radius of Davis-Besse as reported in Table 2.6-1 (of the Environmental Report) and the escalated population in 2040 used as input to the Level 3 Probabilistic Risk Assessment (PRA) is because SECPOP2000 only includes population in the United States. SECPOP2000 calculates estimated population and economic data about any point (specified by longitude and latitude) that lies within the continental United States. The population data in SECPOP2000 are based on 2000 U.S. Census Bureau data. The year 2000 population in a 50-mile radius of Davis-Besse (used as the basis of the escalation) was taken from SECPOP2000. Since SECPOP2000 does not include Canadian population, the 2000

population used in Level 3 PRA underestimated the total population in a 50-mile radius around Davis-Besse. The population data in Table 2.6-1 included the Canadian population. The Level 3 PRA has been revised to include the Canadian population in sectors 30-40 miles/N, 30-40 miles/NNE, 30-40 miles/NE, 40-50 miles/N, 40-50 miles/NNE, and 40-50 miles/NE. The total escalated population for the year 2040 is 2,903,784. The Canadian population is based on the difference of the population reported in Table 2.6-1 and the SECPOP2000 data originally developed.

Section E.3.4.2 of Attachment E of the Environment Report is revised to explain the addition of Canadian population data. Sections E.4.1, E.4.2, E.4.5, and E.9 are revised to reflect the adjusted cost-benefit results. In Section E.10, Table E.3-11 is revised to reflect the Canadian population data. Tables E.3-21 through E.3-32 are revised to reflect the adjusted results of the base case and the sensitivity cases. Tables E.4-1, E.7-2, Table E.7-3, Table E.7-5, and Table E.8-1 are revised to reflect the adjusted cost-benefit results.

In Section E.3.4.2, the statement concerning the declining population related specifically to population estimated from Reference 19 of Attachment E of the Environmental Report; when the population data by year are summed over the counties surrounding Davis-Besse, it shows increasing population until about 2004, and then slightly decreasing population after that until 2008. The population data from Reference 19 are not explicitly provided in Attachment E of the Environmental Report since these data are publicly accessible through the US Census. This observation underscored the conservative assumption of using a constant population escalation factor for each decade through 2040.

See the Enclosure to this letter for the revision to the DBNPS LRA.

Question RAI 4.c

Three SECPOP2000 code errors have been publicized, specifically: 1) incorrect column formatting of the output file, 2) incorrect 1997 economic database file end character resulting in the selection of data from wrong counties, and 3) gaps in the 1997 economic database numbering scheme resulting in the selection of data from wrong counties. Address whether these errors were corrected in the Davis-Besse analysis. If they were not corrected, then provide a revised cost-benefit evaluation of each SAMA with the errors corrected.

RESPONSE RAI 4.c

First Energy Nuclear Operating Company (FENOC) is aware of the code errors reported for SECPOP2000. These code errors, as noted in the request for additional information

(RAI), are unrelated to the population data. For the Davis-Besse Level 3 PRA, only the population data were extracted from SECPOP2000. All other SITE file input parameters were independently developed. Accordingly, there is no need to correct these code errors, nor is there a need to provide a revised cost-benefit evaluation of each SAMA candidate.

Question RAI 4.d

ER Section E.3.4.6.2 does not identify the population base/year reference for the emergency planning zone (EPZ) evacuation speed. Describe how/whether the EPZ evacuation time was corrected for the year 2040 population (and address the population discrepancy noted in RAI 4.b).

RESPONSE RAI 4.d

Reference [4] (in Attachment E of the Environmental Report) does not identify a collection date for the data that were used to estimate the evacuation speed in Section E.3.4.6.2. The evacuation information provided in Reference [4] was assumed to be current as of the 2000 census. However, no correction factor was applied to account for the increased population in 2040 in the original analysis.

Assuming that an increase in population is proportional to a decrease in evacuation speed, the evacuation speed was adjusted from 0.58 meters/second to 0.52 meters/second. This adjustment represents a 9.6 percent decrease in the evacuation speed, which was used to offset a 9.6 percent [$(1.047)^2 = 1.096$] increase in population at the end of the two-decade license renewal period. This decrease in evacuation speed was evaluated as a new sensitivity case (Sensitivity Case E3). The results are provided in Table 4.d-1, below, and show very little change from the base case, indicating that the results are not sensitive to slow evacuation speeds. The base case results shown in Table 4.d-1 includes the updated population (as needed to respond to RAI 4.b); similarly, sensitivity case E3 includes the updated population, to permit an equitable comparison to the base case.

Table 4.d-1: Comparison of Base Case and Case E3

	Internal Events		
	Base	**E3**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	2.30E+00	2.31E+00	0.4%
Economic Impact (50) ($/yr)	1.80E+03	1.80E+03	0.0%

Question RAI 4.e

In ER Section E.3.5.2.3, for Case A1, identify the heat release energy (e.g. thermal, 1 MW) assumed for both the base and sensitivity cases.

RESPONSE RAI 4.e

The energy of release for the base case and sensitivity Case A1 are provided for each release category in Table 4.e-1, below.

Table 4.e-1 Energy of Release: Base Case and Sensitivity Case A1

Release Category	PLHEAT/Energy of Release (watts)	
	Base Case	Sensitivity Case A1
1.1	6.94E+07	2.16E+09
1.2	6.94E+07	2.16E+09
1.3	6.94E+07	2.16E+09
1.4	6.94E+07	2.16E+09
2.1	6.92E+06	6.19E+08
2.2	9.44E+06	6.02E+08
3.1	2.22E+06	2.67E+07
3.2	2.63E+06	1.82E+07
3.3	2.22E+06	2.50E+07
3.4	2.63E+06	1.82E+07
4.1	9.28E+05	1.66E+07
4.2	2.31E+05	1.66E+07
4.3	7.41E+05	1.66E+07
4.4	2.21E+05	1.66E+07
5.1	3.25E+06	2.10E+07
5.2	1.07E+07	6.48E+07
5.3	3.07E+06	1.85E+07
5.4	9.10E+06	5.58E+07
6.1	6.44E+07	2.98E+08
6.2	9.70E+07	4.30E+08
6.3	6.19E+07	3.98E+08
6.4	9.17E+07	4.27E+08
7.1	2.80E+07	1.68E+08
7.2	2.78E+07	1.67E+08
7.3	2.89E+07	1.72E+08
7.4	2.84E+07	1.68E+08
7.5	2.24E+07	1.42E+08
7.6	2.56E+07	1.31E+08
7.7	1.96E+07	1.34E+08
7.8	2.53E+07	1.34E+08
8.1	1.15E+07	1.52E+08
8.2	9.07E+07	5.21E+08
9.1	2.65E+02	2.08E+03
9.2	3.29E+02	2.14E+03

Item 5

Provide the following with regard to the SAMA identification and screening process:

Question RAI 5.a

ER Section E.5.2 describes major contributors to plant CDF, suggested improvements from the IPE study, and specific SAMA candidates identified to address the major contributors and suggested improvements. In addition to the suggested improvements identified in the ER, the IPE (in Section 3, Other Potential Plant Improvements) identifies four potential plant improvements related to the "back-end analysis": 1) BWST level at switchover to sump recirculation, 2) operator actions for inadequate core cooling, 3) emergency plan evacuation criteria, and 4) monitoring of carbon monoxide levels in containment. Describe the status of the implementation of each of these suggested improvements and identify and assess SAMAs to address each unimplemented improvement.

RESPONSE RAI 5.a

In the IPE, Part 6, Section 3, Other Potential Plant Improvements, one insight discussed is borated water storage tank (BWST) refill options. The discussion notes that for some sequences involving steam generator tube ruptures, the BWST inventory could be depleted by injection before the Reactor Coolant System (RCS) was depressurized sufficiently to terminate flow through the broken tube. The discussion also notes that while means are available to provide water to refill the BWST, there is no explicit procedural guidance to taking that step. Since the issuance of the IPE, the EOP has been revised; in EOP Section 8, Steam Generator Tube Rupture, Section 8.54 directs the operators to lineup and transfer the contents of the Clean Waste Receiver Tank (CWRT) to the BWST (if BWST inventory is required). It also directs the operators to procedure DB-OP-06101, "Clean Liquid Radwaste System," which includes specific steps to lineup the CWRT to refill the BWST.

In the IPE, another insight discussed is Operator actions for inadequate core cooling. The discussion notes that different timing of operator inadequate core cooling actions, and particularly those related to RCS depressurization and restarting the RCPs, would have delayed the onset of serious core damage. The discussion also notes that there are concerns regarding the effect of RCP restarts on creep rupture of the SG tubes or RCS for high pressure accidents. Since the IPE, FENOC has prepared Severe Accident Management Guidelines (SAMGs). Davis-Besse SAMG candidate high level actions for all plant damage conditions include the injection of water into the RCS and/or Containment. The likelihood of pressurizer surge line creep rupture, hot leg creep

rupture, and SGTR due to bumping or restarting of the RCPs is addressed for plant conditions which have the primary system pressurized.

In the IPE, another insight discussed is emergency plan evaluation criteria. The discussion notes that a re-examination of evaluation criteria should be accomplished to ensure consistency with the more realistic accident source terms available for severe accidents. On September 30, 2009, Davis-Besse implemented revised Emergency Action Levels (EALs) based on Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels", Revision 5. The NRC approved the revised EALs in a safety evaluation report (DBNPS, Unit 1, Safety Evaluation for Emergency Action Levels (ADAMS Accession number ML083450120)). NEI 99-01 Revision 5 EALs use two isotopic mixes to determine EALs associated with fuel melt and failure. The Davis-Besse station dose assessment program has the ability to perform dose assessment using either mix.

In the IPE, another insight discussed is monitoring of carbon monoxide levels in containment. The discussion notes that if core-concrete interactions occur in a severe accident, significant amounts of flammable carbon monoxide would be generated and consideration of carbon monoxide as well as hydrogen may be appropriate in emergency plan evacuation or severe accident management guidelines. The Davis-Besse SAMGs address hydrogen burn likelihood and resultant containment pressures for various hydrogen concentrations (hydrogen production is assumed to be 50 percent or 75 percent of clad oxidation). Containment pressure change due to core concrete interaction gas evolution is also estimated. The Davis-Besse SAMG Technical Basis Document (TBD) discusses Core Concrete Interactions (CCI), the release of carbon dioxide (CO_2), and the potential for combustible concentrations of carbon monoxide (CO) and hydrogen (H_2) in Containment.

Because the improvements discussed above have been implemented at Davis-Besse, there is no need to identify and assess additional SAMAs.

Question RAI 5.b

ER Section E.5.2 indicates that no plant-specific vulnerabilities that would affect the PRA CDF were identified in the IPEEE. NRC staff notes that the IPEEE safety evaluation report (Section 3.0, of the seismic attachment) states that "The aggregate of the material provided in the submittal and the licensees response to the RAIs is not quite sufficient to meet NUREG 1407" but that "The license did provide an incomplete list of HCLPF values for the plant, with the lowest HCLPF value being 0.26g" and so concluded that the submittal "did come close to meeting the objectives of a focused scope analysis." A FirstEnergy response to an NRC staff RAI on the IPEEE dated May 25, 2000 identifies a number of plant

components with high-confidence low probability of failure (HCLPF) values less than 0.3g:

- **Borated Water Storage Tank roof from sloshing (0.28g)**
- **Masonry Wall No. 2367 associated with 480 V Essential MCC (0.26g)**
- **Masonry Wall No. 3407 associated with Component cooling water room (0.27g)**
- **Masonry Wall No. 4786 associated with Essential Distribution Panel "D2N" (0.27g)**
- **Masonry Wall No. 6107 associated with Control Room Emergency Vent Fan Temperature Switch (0.29g)**

Discuss whether plant improvements to meet 0.3g for these components has been implemented at the plant and, if not, identify and evaluate SAMAs to improve the seismic capacities of each of these components.

RESPONSE RAI 5.b

SAMA SR-01 considers increasing the seismic ruggedness of plant components. As identified in ER Table E.6-1, the Seismic Qualification Utility Group (SQUG) previously identified the need for additional seismic restraints in the plant, and these restraints have been added.

No modifications have been made to the borated water storage tank roof that would increase the seismic capability of the tank roof.

Plant improvements and updated analyses have also been performed on the masonry wall plant components listed that may impact their HCLPF. During the masonry wall project in 2007, changes were made to Masonry wall 3407; the pipe support load was removed from the wall thereby eliminating a major load on the wall. Similarly, changes were made to Masonry wall 6107; the steel beam supporting the wall loads was reinforced. In addition, in the 2006-2007 time frame, the masonry wall analysis was updated for a majority of masonry walls, including Masonry walls 2367 and 4768. The analyses were updated to ensure they met allowable stresses and Design Basis requirements. Although improvements in seismic capacity of the masonry walls have been made, no specific analysis has been performed to determine whether the walls meet the HCLPF value of 0.3g.

In addition, several other SAMAs also meet the intent of improving the seismic capacity of plant components (e.g., AC/DC-01, CC-10, and CW-09).

Question RAI 5.c

None of the SAMA candidates identified in Table E.5-4 appear to be plant-specific SAMAs identified from plant-specific risk insights based on the current PRA model. Clarify how the importance lists were used to develop plant-specific SAMA candidates and justify the apparent absence of any plant-specific SAMA candidates. Also, the basic events identified in importance analysis Tables E.5-2 and E.5-3 are not linked to SAMA candidates. Sections E.5.4 and E.5.5 only discuss the SAMA candidates identified to address basic events with high risk reduction worth (RRW) values. Identify, for each basic event having a RRW benefit value (averted cost risk) greater than the minimum cost of a procedure change at Davis-Besse, the specific SAMA(s) that address each event and describe how the SAMA(s) address the basic event. Identify and evaluate SAMAs for basic events not addressed by an existing SAMA (e.g., flooding related basic events and initiators, including WHAF3ISE, SHAF2ISE, F3AM, and F7L). For any basic event for which no SAMA is identified, provide justification for not identifying a SAMA(s).

RESPONSE RAI 5.c

The final list of SAMA candidates was developed from a combination of generic data, industry SAMA analyses and Davis-Besse-specific insights. The following SAMA candidates were added to the generic list based on Davis-Besse PRA-identified insights:

- SAMA candidate AC/DC-25 (dedicated DC power for AFW) and AC/DC-26 (alternator/generator for turbine-driven auxiliary feedwater (TDAFW) pump) were designed to extend the life of the TDAFW pumps in a station blackout (SBO) event and improve the likelihood of successful restoration of alternating current (AC) power.

- SAMA candidate AC/DC-27 (increased size of SBO fuel oil tank) was also designed to help mitigate an SBO event.

- SAMA candidate CB-21 (pressure sensors between the two in-series Decay Heat Removal (DHR) System suction valves) was designed to help reduce the likelihood of ISLOCA events.

- SAMA candidate CC-19 (automatic switchover of high pressure injection (HPI) and low pressure injection (LPI) suction from the BWST to the containment sump) was designed to increase the reliability of the switchover during a loss of coolant accident (LOCA) event.

- SAMA candidate CC-20 (modify hardware and procedures to allow using make-up pumps for high pressure recirculation from the containment sump) was

designed improve the reliability of high pressure recirculation following the loss of HPI.

- SAMA candidate CC-21 (reduce the BSWT level at which switchover to containment recirculation is initiated) was designed to extend the time available to accomplish BWST refill..

- SAMA candidate CP-19 (install a redundant containment fan system) was designed to increase containment heat removal ability. This SAMA candidate was added as a variation to CP-18 to provide a redundant containment cooling function, in the form of containment fan coolers.

- SAMA candidates CW-24 (adding a diversified CCW pump) and CW-25 (providing the capability to cool makeup pumps with fire water on loss of CCW) were designed to mitigate the total loss of CCW cooling.

- SAMA candidate FW-16 (surveillance of manual AFW suction valves) was designed to improve the reliability of alternate sources of AFW water supply.

- SAMA candidate HV-06 (procedure guidance for alternate means of switchgear cooling) was designed to prevent the loss of one train of service water in the event of loss of one HVAC fan for the service water pump room. This SAMA candidate was developed from Davis-Besse IPE insights.

Evaluating Basic Events with Potential Benefit Greater Than the Cost of a Procedure Change

The internal events and LERF basic events with an RRW value estimated to be equal to or greater than the cost of a procedure change were evaluated. These basic events were dispositioned by either identifying resulting SAMAs or presenting the reason for no new SAMA candidate. One new SAMA candidate (OT-9R) resulted from this evaluation.

An estimate of the cost-benefit versus RRW was developed for the internal events basic events calculated for the base PRA model. The minimum cost of a procedure change was assumed to be $10,000. In addition, the minimum cost of a hardware modification was estimated to be $100,000. The cost-benefit versus RRW assumed that cost-benefit was directly proportional to the reduction in core damage frequency (CDF). Cost is not perfectly correlated with CDF, due to the fact that different scenarios, even with the same CDF, will result in different distributions of release categories. It is judged, however, that this correlation provides a reasonable estimate of potential benefit along with what is judged to be a low cost for a procedure change, and provides strong confidence that cost-effective SAMA candidates will be captured.

For the total benefit for the hazard group (Bt), the cost-benefit versus RRW used the maximum derived benefit of $349,147.

The following formula is used for deriving the estimated benefit by hazard group based on RRW:

$$EB_{(BE)} = B_t\left(1-\frac{1}{RRW}\right)$$

where,

$EB_{(BE)}$ = the estimated benefit based on a basic event

B_t = the total benefit for the hazard group (internal events, fire, or seismic)

RRW = the RRW for the basic event from the PSA, by hazard, assuming the basic event failure probability is reduced to zero.

The RRW for the Level 2 PRA basic events may be calculated based on LERF rather than CDF. Additional conservatism is added by treating Level 2 PRA basic event RRW values based on LERF as if they were based on CDF (i.e., the use of Bt significantly overstates their benefit), and the degree of conservatism could be large.

Based on these estimates, an RRW value of 1.03 was calculated to have a maximum cost benefit of $10,000 and an RRW of 1.40 was estimated to have a maximum cost benefit of $100,000. The maximum cost benefit is based on the RRW of the basic event being reduced to 1.0 (basic event modeled as perfect). For all basic events having an RRW value estimated to be at, or above, the value of a procedure change, a disposition was provided either identifying the SAMA candidate(s) addressing that basic event or a description as to why the basic event was not addressed in a SAMA candidate. No basic events had an RRW value equal to, or greater than the estimated cost of a hardware modification. Table 5.c-1, below, lists the basic events with the highest RRW for CDF.

Table 5.c-2, below, tabulates the basic events with the highest RRW for LERF. The estimated benefit for each basic event was derived by taking the RRW for LERF and applying the maximum total benefit used for the CDF basic events. This is very conservative, since the total maximum benefit does not apply only to LERF. For all basic events having an RRW value estimated to be at, or above, the value of a procedure change, a disposition was provided either identifying the SAMA candidate(s) addressing that basic event or a description as to why the basic event was not addressed in a SAMA candidate. No basic events had an RRW value equal to, or greater than the estimated cost of a hardware modification.

Basic events WHAF3ISE, SHAF2ISE, F3AM, and F7L did not have RRW values with potential benefit equal to, or greater than, the minimum cost of a procedure change. Basic event F7L, a large circulating water flood in the Turbine Building, did, however, result in an RRW value greater than the minimum cost of a procedure change for the 95 percent uncertainty CDF model. SAMA candidate FL-01 (improve inspection of rubber expansion joints on main condenser) was initially identified to address basic event F7L, and was designed to reduce the frequency of a large circulating water system flooding event due to failure of the circulating water system expansion joints. Based on the F7L RRW value from the 95 percent uncertainty CDF model and its original screening of "Very Low Benefit," SAMA candidate FL-01 was reevaluated and screened as "Already Implemented," as discussed in the response to RAI 6.k.

The ER is revised (numerous locations) to identify that there are now 168 SAMA candidates that were evaluated instead of the original 167. Also, ER Table E.5-4 is revised to include changes identified in Tables 5.c-1 and 5.c-2, below.

See the Enclosure to this letter for the revision to the DBNPS LRA.

Table 5.c-1 – Basic Event Level 1 PRA Importance				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
UHAMUHPE	2.59E-01	1.349	Failure to initiate makeup/HPI cooling after loss of all feedwater	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.
QHAMDFPE	2.45E-01	1.324	Failure to start MDFP after loss of feedwater	SAMA candidate FW-17R evaluates implementing an automatic start of the motor-driven feed pump (MDFP) on loss of main feedwater (MFW).
QHARCPCE	2.32E-01	1.302	Operators fail to trip RCPs after a total loss of seal cooling	SAMA candidate CW-26R evaluates implementing an automatic RCP trip on high bearing cooling temperature or loss of CCW flow to the RCP thermal barrier cooler and loss of seal injection flow.
T3	1.96E-01	1.243	LOOP (initiating event)	Numerous SAMA candidates that address LOOP were evaluated: AC/DC-01, additional battery capacity AC/DC-14, install gas turbine generator AC/DC-25, provide dedicated DC system to TDAFW pumps AC/DC-26, provide alternator/generator driven by TDAFW pumps AC/DC-27, increase SBO fuel oil tanks size
EHASBDGE	1.64E-01	1.196	Operators fail to align power from SBO diesel generator to supply MDFP	SAMA candidate AC/DC-28R evaluates the automatic start of the SBO diesel and loading to Bus D2 upon loss of power to Bus D2.

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
EHASBD1E	1.58E-01	1.187	Operators fail to start SBO diesel generator and align to bus D1	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.
EHAD2DGE	1.53E-01	1.181	Operators fail to align power from EDG 1-1 or EDG 1-2 to supply MDFP given LOOP	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
T1	1.35E-01	1.156	Reactor/turbine trip (initiating event)	This is based on a somewhat conservative T1 value of 1.02/yr. Davis-Besse trip occurrence frequency is considered representative of industry values. SAMA candidates have been evaluated that address various Davis-Besse important scenarios following a reactor/turbine trip. CC-01, evaluates the installation of an independent active or passive HPI system. CW-26R, evaluates an automatic RCP trip on high motor bearing temperature or loss of CCW flow to the RCP thermal barrier cooler and loss of seal injection flow. FW-17R, evaluates an automatic start of the motor driven feedwater pump. HV-01, evaluates a redundant train for ventilation. HV-03, evaluates the staging of backup fans in the switchgear room.
QHAOVF2E	1.22E-01	1.139	Operators fail to take local manual control of TDAFW pump 1-2 speed.	SAMA candidate AC/DC-25 provides a dedicated DC system to TDAFW pumps and SAMA candidate AC/DC-26 provides an alternator/generator driven by TDAFW pumps. These SAMA candidates would eliminate the need for local manual control of the TDAFW pumps.
ZHARCPCE	1.10E-01	1.124	Operators fail to trip RCPs following loss of seal cooling	SAMA candidate CW-26R evaluates implementing an automatic RCP trip on high bearing cooling temperature or loss of CCW flow to the RCP thermal barrier cooler and loss of seal injection flow.

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
WHASPREE	1.07E-01	1.12	Failure to recover CCW using spare CCW train (prior to damage)	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.
QMBAFP11	7.61E-02	1.082	AFW Train 1 in maintenance	This estimated benefit of this basic event is below the minimum estimated cost of a hardware modification. The following SAMA candidates address improvements to the reliability of AFW in loss of off-site power scenarios: AC/DC-25, provide dedicated DC system to TDAFW pumps AC/DC-26, provide alternator/generator driven by TDAFW pumps
XHOS-CCW1RUN2STBY	7.54E-02	1.082	CCW Pump 1 running, Pump 2 in standby	This is a plant configuration probability in the model. It does not contribute to risk.
EDG0012F	7.12E-02	1.077	EDG 1-2 fails to run	SAMA candidate AC/DC-14 evaluates adding a gas turbine generator as an additional source of on-site power.
ZOP007BR	7.09E-02	1.076	Failure to restore off-site power	Numerous SAMA candidates that address LOOP were evaluated: AC/DC-01, additional battery capacity AC/DC-14, install gas turbine generator AC/DC-25, provide dedicated DC system to TDAFW pumps AC/DC-26, provide alternator/generator driven by TDAFW pumps AC/DC-27, increase SBO fuel oil tanks size

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
TMPP43XF-CC_ALL	6.79E-02	1.073	All CCW pumps fail to run due to CCF (initiating event)	SAMA candidate CW-24 evaluates the standby CCW pump with a pump diverse from the other two CCW pumps.
XHOS-CCW2RUN1STBY	6.57E-02	1.07	CCW Pump 2 running, Pump 1 in standby	This is a plant configuration probability in the model. It does not contribute to risk.
R	6.37E-02	1.068	SGTR (initiating event)	Davis-Besse is scheduled to install new steam generators in 2013. This modification, with resulting reduction in SGTR frequency, is not reflected in the current PRA model. This plant improvement is assumed to result in a reduction risk importance of SGTR events.
EHAD1ACE	5.90E-02	1.063	Failure to lineup alternate source to D1	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.
T2	5.86E-02	1.062	Plant trip due to loss of MFW (initiating event)	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
NORCVRT3	5.57E-02	1.059	Offsite power recovery not possible after a tornado.	Numerous SAMA candidates that address LOOP were evaluated: AC/DC-01, additional battery capacity AC/DC-14, install gas turbine generator AC/DC-25, provide dedicated DC system to TDAFW pumps AC/DC-26, provide alternator/generator driven by TDAFW pumps AC/DC-27, increase SBO fuel oil tanks size

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
AV	5.12E-02	1.054	Reactor vessel rupture	Reactor vessel rupture is a low probability event that that is assumed to result in guaranteed core damage. No applicable SAMA candidates were considered possible to prevent core damage.
QTP000XA-CC_1_2	5.13E-02	1.054	CCF of two components: QTP0001A & QTP0002A (TDAFW)	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
QTP0001A	4.90E-02	1.051	AFP/T-1 fails to start	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
QMBAFP12	4.67E-02	1.049	AFW Train 2 in maintenance	The estimated benefit for this basic event is below the cost of a hardware modification. The following SAMA candidates address improvements to the reliability of AFW in LOOP scenarios: AC/DC-25, provide dedicated DC system to TDAFW pumps AC/DC-26, provide alternator/generator driven by TDAFW pumps
ZOP006FR	4.58E-02	1.048	Failure to restore off-site power	Numerous SAMA candidates that address LOOP were evaluated: AC/DC-01, additional battery capacity AC/DC-14, install gas turbine generator AC/DC-25, provide dedicated DC system to TDAFW pumps AC/DC-26, provide alternator/generator driven by TDAFW pumps AC/DC-27, increase SBO fuel oil tanks size

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
S	4.35E-02	1.045	Small LOCA (initiating event)	SAMA candidate CC-01 evaluates the installation of an independent active or passive HPI system. SAMA candidate CC-19 evaluates the implementation of automatic switchover of HPI and LPI suction from the BWST to the to containment sump for LOCAs.
T13A-1-3-IEF	4.18E-02	1.044	Loss of CCW Train 1 initiating event Pump 1 running	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
MHARMVTE	4.17E-02	1.043	Operators fail to compensate for loss of room cooling for makeup pumps.	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.
XHAMUCDE	4.10E-02	1.043	Operators fail to attempt cooldown via makeup/HPI cooling.	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.
T13A-2-3-IEF	3.93E-02	1.041	Loss of CCW Train 2 initiating event Pump 2 running	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
EMBEDG12	3.85E-02	1.04	EDG Train 2 in maintenance	SAMA candidate AC/DC-14 evaluates adding a gas turbine generator as an additional source of on-site power.

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
CHASGDPE	3.63E-02	1.038	Operators fail to cooldown during a SGTR	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R. Also, Davis-Besse is scheduled to install new steam generators in 2013. This modification, with resulting reduction in SGTR frequency, is not reflected in the current PRA model. This plant improvement is assumed to result in a reduction risk importance of SGTR events.
FMFWTRIP	3.71E-02	1.038	MFW/ICS faults following trip	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
FMM00003	3.52E-02	1.037	Any MSSVs on SG1 fail to reseat	SAMA candidate CB-22R evaluates the use of a "gagging device" to close a stuck open MSSV.
EDG0012A	3.46E-02	1.036	EDG 1-2 fails to start	SAMA candidate AC/DC-14 evaluates adding a gas turbine generator as an additional source of on-site power.
AASGTR11	3.42E-02	1.035	SGTR occurs on OTSG 1-1 (split fraction)	Davis-Besse is scheduled to install new steam generators in 2013. This modification, with resulting reduction in SGTR frequency, is not reflected in the current PRA model. This plant improvement is assumed to result in a reduction risk importance of SGTR events.

Table 5.c-1 – Basic Event Level 1 PRA Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
LHAMSIVE	3.34E-02	1.035	Failure to close MSIV and isolate steam generator containing ruptured tube	Davis-Besse is scheduled to install new steam generators in 2013. This modification, with resulting reduction in SGTR frequency, is not reflected in the current PRA model. This plant improvement is assumed to result in a reduction risk importance of SGTR events.
QHAMDF3E	3.34E-02	1.035	Failure to start MDFP prior to depletion of BWST during makeup	SAMA candidate FW-17R evaluates implementing an automatic start of the motor-driven feed pump (MDFP) on loss of main feedwater (MFW). SAMA candidate CC-22R evaluates implementing an automatic refilling of the BWST.
QTP0002A	3.25E-02	1.034	AFP/T-2 fails to start	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
EDG0011F	3.13E-02	1.032	EDG 1-1 fails to run	SAMA candidate AC/DC-14 evaluates adding a gas turbine generator as an additional source of on-site power.
FCIRCTMP	3.00E-02	1.031	Circ water temperature not acceptable	This is a PRA model flag. It is not a candidate for a SAMA. No SAMA candidate considered.
RRW of 1.03 is estimated to have a cost of approximately $10,000. This is assumed to be the minimum cost of a procedure change.				

Table 5.c-2 – Basic Event LERF Importance				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
R	9.00E-01	10.048	SGTR (initiating event)	Davis-Besse is scheduled to install new steam generators in 2013. This modification, with resulting reduction in SGTR frequency, is not reflected in the current PRA model. This plant improvement is assumed to result in a reduction risk importance of SGTR events.
XHAMUCDE	6.10E-01	2.563	Operators fail to attempt cooldown via makeup/HPI cooling	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.
CHASGDPE	5.40E-01	2.175	Operators fail to cooldown during a SGTR	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.
LHAMSIVE	4.97E-01	1.989	Failure to close MSIV and isolate steam generator containing ruptured tube	A SAMA candidate was developed that presents the highest worth PRA human actions to the Davis-Besse operator training. SAMA candidate OT-09R was added to the initial list of SAMA candidates, but subsequently found to be already implemented at Davis-Besse.

Table 5.c-2 – Basic Event LERF Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
AASGTR11	4.81E-01	1.926	SGTR occurs on OTSG 1-1 (split fraction)	Davis-Besse is scheduled to install new steam generators in 2013. This modification, with resulting reduction in SGTR frequency, is not reflected in the current PRA model. This plant improvement is assumed to result in a reduction risk importance of SGTR events.
AASGTR12	3.93E-01	1.646	SGTR occurs on OTSG 1-2 (split fraction)	Davis-Besse is scheduled to install new steam generators in 2013. This modification, with resulting reduction in SGTR frequency, is not reflected in the current PRA model. This plant improvement is assumed to result in a reduction risk importance of SGTR events.
RRW of 1.40 is estimated to have a cost of approximately $100,000. This is assumed to be the minimum cost of a hardware modification.				
FMM00003	7.90E-02	1.086	Any MSSVs on SG1 fail to reseat	SAMA candidate CB-22R evaluates the use of a "gagging device" to close a stuck open MSSV.
VD-IEF	7.54E-02	1.082	ISLOCA due to internal rupture of DHR suction valves	SAMA candidate CB-21 evaluates placing pressure measurements between the two DHR suction valves in the RCS hot leg allowing early detection of inboard isolation valve leakage.
FLCO101F	7.31E-02	1.079	Logic card fails during operation – MSIV 101 fails to close	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.

Table 5.c-2 – Basic Event LERF Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
LPPNISOZ	7.18E-02	1.077	ISLOCA occurs in non-isolable portion of DHR system	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
FMM00004	6.80E-02	1.073	Any MSSVs on SG2 fail to reseat	SAMA candidate CB-22R evaluates the use of a "gagging device" to close a stuck open MSSV.
FLC0100F	6.13E-02	1.065	Logic card fails during operation – MSIV 100 fails to close	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
QHAMDFPE	5.96E-02	1.063	Failure to start MDFP as backup to turbine-driven feedwater pumps for transient, Small LOCA or SGTR events	SAMA candidate FW-17R evaluates implementing an automatic start of the motor-driven feed pump (MDFP) on loss of main feedwater (MFW).
EC1ZXXXN-CC_1_2	5.19E-02	1.055	CCF of two components: EC1Z089N & EC1Z100N	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
LPSRC2BH	4.93E-02	1.052	Press switch PSH RC2B4 fails high – fails DHR	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
LPSZ416H	4.93E-02	1.052	Press switch PSH 7531A fails high - fails DHR	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.

Table 5.c-2 – Basic Event LERF Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
LMVF012R	4.53E-02	1.047	Internal rupture of DH 12 (annual frequency)	SAMA candidate CB-21 evaluates placing pressure measurements between the two DHR suction valves in the RCS hot leg allowing early detection of inboard isolation valve leakage.
LMBCWRT1	4.12E-02	1.043	CWR Train 1 unavailable due to maintenance	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
EDG0012F	3.47E-02	1.036	EDG 1-2 fails to run	SAMA candidate AC/DC-14 evaluates adding a gas turbine generator as an additional source of on-site power.
FCIRCTMP	3.00E-02	1.031	Circ water temperature not acceptable	This is a PRA model flag. It is not a candidate for a SAMA. No SAMA candidate considered.
FVV011BT	3.04E-02	1.031	AVV ICS11B fails to reseat after steam	The estimated benefit for this basic event is below the cost of a hardware modification. No SAMA candidate considered.
LMVF011R	3.01E-02	1.03	Internal rupture of DH 11 (annual frequency)	SAMA candidate CB-21 evaluates placing pressure measurements between the two DHR suction valves in the RCS hot leg allowing early detection of inboard isolation valve leakage.

Table 5.c-2 – Basic Event LERF Importance (continued)				
Event Name	**F-V**	**RRW**	**Description**	**Disposition**
ELOOPRT	2.93E-02	1.03	LOOP given reactor trip	Numerous SAMA candidates that address LOOP were evaluated: AC/DC-01, additional battery capacity AC/DC-14, install gas turbine generator AC/DC-25, provide dedicated DC system to TDAFW pumps AC/DC-26, provide alternator/generator driven by TDAFW pumps AC/DC-27, increase SBO fuel oil tanks size
RRW of 1.03 is estimated to have a cost of approximately $10,000. This value is assumed to be the minimum cost of a procedure change.				

Question RAI 5.d

ER Section E.5.3, E.5.4, and E.5.5 discuss significant contributors to core damage frequency (CDF) and large early release frequency (LERF). These sections and the associated tables show that there are a number of operator errors and non-recovery actions that occur in these listings, but report that no weaknesses in training or procedures were identified. Given: 1) the significant number of operator errors in these lists, 2) that human errors are among the most dominant failure modes presented in the importance Tables E.5-2 (i.e., the first 9 basic events listed by RRW are human error events) and E.5-3, and 3) that operator errors often have relatively high failure probabilities, provide the following:

i. **Explain the process used to make the determination that there were no opportunities to improve procedures and training.**

ii. **Discuss whether any of the risk significant operator action failures could be addressed by a SAMA to automate the function (i.e., automating tripping of the RCPs after a loss of seal cooling -see RAI 7.a).**

RESPONSE RAI 5.d

5.d.i

The Human Failure Events (HFEs) included in the dominant cutsets, and identified in the Level 1 and LERF importance tables (as discussed in ER Sections E.5.3, E.5.4 and E.5.5) were reviewed. In the Davis-Besse PRA, the EPRI software supporting the Computer-Aided Fault Tree Analysis (CAFTA) Software, the Human Reliability Analysis (HRA) Calculator, was utilized to quantify and document the HRA analysis. The documentation for each HFE includes a discussion of the action, associated cues, relevant procedures, training, assumptions, staffing, performance shaping factors, and timing. The review concluded that adequate procedures and training were in place; no specific weaknesses were identified in the review of the HFEs.

By their nature, and the way in which they support system fault trees and functional event trees, operator actions are recognized as a key source of model uncertainty and important contributors to core damage. Accordingly, operator actions are discussed in ER Sections E.5.3, E.5.4, and E.5.5. Over the last fifteen years, there has been a significant industry effort in improving procedure content, procedure use, human error reduction techniques, and training.

5.d.ii

In addition to the new SAMAs addressed in RAI 7, two additional SAMA candidates were evaluated to address automating risk significant operation actions: SAMA candidate AC/DC-28R (automatically start and load the SBODG on Bus D2 upon loss of power to the bus), and SAMA candidate OT-08R (automatically start and load the SBODG on Bus D2 upon loss of power to the bus in combination with automatically starting the MDFP). Table 5.d-1 and Table 5.d-2, below, provide the internal event and total benefit results for SAMA candidates AC/DC-28R and OT-08R, respectively. Table 5.d-3, below, provides the final results for the ten sensitivity cases for SAMA candidate AC/DC-28R and OT-08R. The implementation cost for SAMA candidate AC/DC-28R was estimated as $1,600,000. Therefore, this SAMA candidate is not cost-beneficial at Davis-Besse. The implementation cost for SAMA candidate OT-08R was estimated as $4,400,000. Therefore, this SAMA candidate is not cost-beneficial at Davis-Besse.

Table 5.d-1: Internal Events Benefit Results for SAMA Candidates AC/DC-28R and OT-08R

Case	**AC/DC-28R** (Auto SBODG)	**OT-08R** (Auto SBODG & MDFP)
Off-site Annual Dose (rem)	2.23E+00	2.10E+00
Off-site Annual Property Loss ($)	1.74E+03	1.63E+03
Comparison CDF	1.0E-05	1.0E-05
Comparison Dose (rem)	2.30E+00	2.30E+00
Comparison Cost ($)	1.80E+03	1.80E+03
Enhanced CDF	8.3E-06	5.7E-06
Reduction in CDF	**17.00%**	**43.00%**
Reduction in Off-site Dose	**3.04%**	**8.70%**
Immediate Dose Savings (On-site)	$138	$348
Long Term Dose Savings (On-site)	$600	$1,518
Total Accident Related Occupational Exposure **(AOE)**	$738	$1,866
Cleanup/Decontamination Savings (On-site)	$22,502	$56,916
Replacement Power Savings (On-site)	$22,766	$57,584
Averted Costs of On-site Property Damage **(AOSC)**	$45,267	$114,500
Total On-site Benefit	**$46,005**	**$116,366**
Averted Public Exposure **(APE)**	$1,718	$4,908
Averted Off-site Damage Savings **(AOC)**	$736	$2,086
Total Off-site Benefit	**$2,454**	**$6,994**
Total Benefit (On-site + Off-site)	**$48,459**	**$123,360**

Table 5.d-2: Total Benefit Result for SAMA Candidates AC/DC-28R and OT-08R

	AC/DC-28R (Auto_SBODG)	**OT-08R** (Auto_SBODG & MDFP)
Internal Events	$48,459	$123,360
Fires, Seismic, Other	$222,912	$567,455
Total Benefit	$271,371	$690,815

Table 5.d-3: Final Results of the Sensitivity Cases for SAMA Candidates AC/DC-28R and OT-08R

SAMA ID	Repair Case	Low Discount Rate Case	High Discount Rate Case	On-site Dose Case	On-site Clean-up Case
AC/DC-28R	$169,380	$409,899	$187,033	$275,551	$313,374
OT-08R	$432,838	$1,043,605	$476,456	$701,388	$797,058

SAMA ID	Replacement Power Case	Multiplier Case	Evacuation Speed	Off-site Economic Cost	95th CDF Case
AC/DC-28R	$356,944	$387,673	$302,292	$272,745	$393,488
OT-08R	$907,264	$986,879	$721,735	$692,189	$1,001,682

Question RAI 5.e

Table E.5-2 identifies events QMBAFP11 and QMBAFP12 representing unavailability of Auxiliary Feedwater (AFW) Trains 1 and 2, respectively, due to maintenance. Provide an evaluation of a SAMA to improve the availability of the AFW pumps by making improvements to maintenance practices or by making hardware modifications.

RESPONSE RAI 5.e

The events QMBAFP11 and QMBAFP12 represent unavailability of AFW trains 1 and 2. The AFW maintenance unavailability data in the PRA is based on the Maintenance Rule data. The SAMA PRA model includes the following: AFW train 1 in maintenance 285 hours and AFW train 2 in maintenance 311 hours, over 24,209 hours (3 years). These values equate to a maintenance unavailability of 1.18E-2/yr and 1.29E-2/yr for AFW trains 1 and 2, respectively. This data is consistent with the generic Industry unavailability data in NUREG/CR-6928 for a turbine-driven AFW pump of 5.44E-3/yr. Improvements to maintenance practices are proposed and evaluated as a normal course of business to maintain AFW train unavailability at its lowest achievable value. Safety-related hardware modifications are costly, and, based on the industry unavailability data, a SAMA to improve the availability of the AFW pumps is not expected to be cost-beneficial.

Question RAI 5.f

Table E.5-4 does not provide the source for identifying SAMAs CC-19, CW-24, and CW-25. ER Section E.5.2 implies that CW-24 and CW-25 were identified to address IPE risk insights. Clarify the basis for identifying these SAMA candidates.

RESPONSE RAI 5.f

The basis for identifying SAMA candidates CC-19, CP-19, CW-24 and CW-25 were inadvertently omitted from Table E.5-4. The following provides a discussion of the basis for each of these SAMA candidates.

CC-19: Davis-Besse currently has the automatic switchover of HPI and LPI suction from the BWST to the containment sump removed. SAMA candidate CC-19 examined re-installing the automatic switchover of HPI and LPI suction from the BWST to the containment sump. The first MLOCA cutset (cutset #12) included basic event ZHALPRME (operators fail to initiate low pressure recirculation) as a single-element cutset.

CP-19: This SAMA candidate evaluates the installation of a redundant containment fan system. SAMA candidate CP-18 was taken from the generic list of SAMA candidates, and evaluates the implementation of a redundant containment spray system. SAMA candidate CP-19 was added as a variation to CP-18 to provide a redundant containment cooling function, in the form of containment fan coolers.

CW-24: This SAMA candidate to add a diversified CCW pump was developed based on the high importance of CCW, as indicated in cutsets and RRW importance values.

CW-25: This SAMA candidate to provide the ability to cool makeup pumps using fire water in the event of loss of CCW was developed based on the high importance of CCW, as indicated in cutsets and RRW importance values.

ER Table E.5-4, "List of Initial SAMA Candidates," rows CC-19, CP-19, CW-24 and CW-25, are revised to include a reference source.

See the Enclosure to this letter for the revision to the DBNPS LRA.

Question RAI 5.g

Several SAMA candidates identified in Table E.6-1 are subsumed in another SAMA candidate (e.g., AC/DC-06, AC/DC-09, AC/DC-20). For each subsumed SAMA candidate, provide an assessment of its implementation cost relative to that of the SAMA into which it was subsumed. If the implementation cost of the subsumed SAMA is less, provide a revised basis for the Phase I screening and Phase II cost-benefit evaluation if it meets Criterion F.

RESPONSE RAI 5.g

SAMA candidate CB-08 was subsumed in SAMA candidate CB-07 in Table E.6-1. SAMA candidate CB-07 was screened as already been implemented at Davis-Besse. The nature of the operation action/training is similar in both SAMA candidates. Therefore, SAMA candidate CB-08 was re-screened as Criterion B (Already Implemented). Accordingly, there was no need to determine the cost of implementation and assess the cost-benefit of SAMA candidate CB-08. ER Table E.6-1 is revised to identify the re-screening of SAMA candidate CB-08.

The SAMA candidates subsumed in Phase I (AC/DC-06, AC/DC-09, AC/DC-20, and CC-08) have an equivalent or higher cost of implementation than the SAMA candidates evaluated in Phase II. Nonetheless, an analysis was performed to assess the cost-benefit of the subsumed SAMA candidates. The total benefit was derived from the SAMA candidates into which they were subsumed and compared to the cost of implementation. Table 5.g-1 provides the results of the cost-benefit evaluation. None of the subsumed SAMA candidates are cost-beneficial to implement at Davis-Besse.

See the Enclosure to this letter for the revision to the DBNPS LRA.

Table 5.g-1: Final Results of the Cost-Benefit Evaluation for Subsumed SAMA Candidates

SAMA ID	Modification	Estimated Benefit	Cost Estimate	Conclusion
AC/DC-06	Provide additional DC power to the 120/240V vital AC system.	$94,363	$1,750,000	Not Cost Effective
AC/DC-09	Provide an additional diesel generator.	$94,363	$2,800,000	Not Cost Effective
AC/DC-20	Add a new backup source of diesel generator cooling.	$33,745	$700,000	Not Cost Effective
CC-08	Add the ability to automatically align ECCS to recirculation mode upon BWST depletion.	$15,155	$1,500,000	Not Cost Effective

Question RAI 5.h

A few SAMA candidates identified in Table E.6-1 are screened for Very Low Benefit based on low contribution to LERF (e.g., CB-02, CP-21, OT-07). The ER does not provide sufficient information to assess the contribution of LERF to population dose-risk and offsite economic cost-risk relative to the total contribution from all release categories. Considering that the benefit of a SAMA is potentially based on the contribution from multiple release categories, provide additional justification for screening these SAMAs on Very Low Benefit.

RESPONSE RAI 5.h

SAMA candidate CB-02 addresses the reliability of containment isolation, and was included in the generic SAMA list within the CB (containment bypass) category. Isolation failure leads to a LERF event. Therefore, this SAMA candidate has no impact on CDF. At Davis-Besse, isolation failure is not a significant contributor to LERF, based on LERF basic event RRW values. Improving containment isolation reliability will not have any significant improvement in other release categories; therefore this SAMA candidate was not considered further.

SAMA candidate CP-21 addresses installing a passive hydrogen control system. A hydrogen burn or detonation typically leads to an early large release. A hydrogen burn or detonation is not risk-significant for LERF at Davis-Besse; therefore this SAMA candidate was not considered further.

SAMA candidate OT-07 is designed to reduce the likelihood of a main steam line break upstream of the main steam isolation valves (MSIVs). This SAMA candidate should not have been eliminated based on LERF. Rather, main steam line breaks are not a significant contributor to either CDF or LERF since they are not found in the top 100 cutsets or the list of either Level 1 or Level 2 risk-significant basic events. The disposition of this SAMA in ER Table E.6-1, "Qualitative Screening of SAMA Candidates," is revised to include a reference to CDF.

See the Enclosure to this letter for the revision to the DBNPS LRA.

Question RAI 5.i

SAMA CB-18, "direct steam generator flooding after a steam generator tube rupture (SGTR), prior to core damage," was screened in Table E.6-1 because it could impact efforts to mitigate the SGTR. This SAMA was determined to be potentially cost-beneficial in previous SAMA analyses (e.g., Diablo Canyon, TMI-1). Provide a cost-benefit evaluation of this SAMA.

RESPONSE RAI 5.i

In the Davis-Besse PRA model, steam generator tube rupture sequences resulting in core damage are placed in one of the following core damage bins: RRY, RRN, RIY, or RIN. Core damage bins RRN and RIN represent sequences in which feedwater is unavailable to the steam generators. In these sequences, it would be impossible to flood the steam generators because no feedwater is available to do so. For core damage bins RRY and RIY, feedwater is available, and it was judged that scrubbing would occur in the steam generator. The auxiliary feedwater nozzles spray high into the tubes and would be expected to provide scrubbing even if the break location was not flooded. Therefore, flooding the steam generators as suggested in CB-18 provides no additional scrubbing benefit, and as such, a cost-benefit evaluation of those SAMAs is not warranted.

Item 6

Provide the following with regard to the Phase II cost-benefit evaluations:

Question RAI 6.a

ER Section E.7.2 states that an expert panel developed the implementation cost estimates for each of the SAMAs. Briefly, describe the level of detail used to develop the cost estimates (i.e., the general cost categories considered). Also, clarify whether the cost estimates accounted for inflation, contingency costs associated with unforeseen implementation obstacles, replacement power during extended outages required to implement the modifications, and maintenance and surveillance costs during plant operation.

RESPONSE RAI 6.a

The Expert Panel process was a collegial review process that relied upon the expertise and judgment of long-term site staff drawn from engineering, operations, procurement, and project management, and assisted by select support personnel (License Renewal, SAMA & probabilistic risk assessment (PRA)). The Panel reviewed each SAMA candidate and, based on their professional expertise and judgment, approximated the costs associated with implementation processes and equipment.

Main cost categories considered included:

- equipment, including the specific mechanical or electrical components identified in the SAMA (e.g., gas turbine-powered generator), and associated piping and piping components, and electrical cables, switchgear, connectors and conduit;
- fuel (natural gas or petroleum-based fuels), if appropriate;
- space requirements, and whether existing space was available or new spaces need to be constructed to house and protect the equipment or for storage of associated fuel and supporting equipment; and,
- extent of modifications, considering whether modifications were safety-related (higher costs) or nonsafety-related, the seismic requirements (higher costs), calculation requirements (higher costs), whether piping or electrical runs would be required between structures or through walls (higher costs), or whether the Control Room envelope was potentially impacted (higher costs).

Some implementation costs were assigned a standard value based upon plant experience or estimated man-hours required:

- minimal procedure changes will be between $10,000 and $50,000;
- procedure changes with Engineering support will be between $50,000 and $200,000;
- procedure changes with Engineering support and testing or training required will be between $200,000 and $300,000; and,
- minimal physical plant changes (modifications) start at $100,000.

Least cost "out-of-the-box" options were included wherever possible (e.g., securing retail store small generator(s)). Detailed design concepts were not developed by the Expert Panel, but every effort was made to identify and reasonably price all activities that need to be performed in support of each SAMA candidate (i.e., "conceptually estimated," as described by NEI 05-01, "Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document," (Nov. 2005), Section 7.2, "Cost of SAMA Implementation"). These support activities included costs associated with procurement, installation, long-term maintenance, surveillance, calibration, and initial and ongoing training. Inflation, contingency costs associated with unforeseen implementation obstacles, and replacement power costs during extended outages required to implement modifications were not specifically identified or included in the cost estimates.

Question RAI 6.b

SAMA CC-19, "provide automatic switch over of HPI and LPI suction from the BWST to containment sump for LOCAs," has an estimated implementation cost of $1.5M. Table E.6-1 states that Davis-Besse already has this capability but that the feature has been deactivated, and that the cost would be minor to reactivate this feature. The estimated cost of $1.5M seems very high based on this description. Furthermore, other SAMA analyses have estimated the cost of this SAMA to range from $265K (Robinson) to $1M (Catawba). Provide a more detailed description of this modification and justification for the estimated cost.

RESPONSE RAI 6.b

The SAMA Expert Panel made the following assumptions regarding SAMA candidate CC-19 to provide automatic switchover of HPI and LPI suction from the BWST to the containment sump:

- the hardware for automatic switchover is already in-place, but not connected, so reconnection and reactivation of the equipment is necessary;
- the associated valves were de-powered in support of Appendix R criteria;
- Appendix R analyses would need to be re-performed (approximately $500K);
- the change would require a safety-related modification due to the safety-significance of the affected equipment, and calculation support would be necessary (approximately $500K);
- procedure changes with Engineering support and initial testing or training required (approximately $300K); and,
- ongoing testing, surveillances, maintenance and training (approximately $200K).

Estimated cost to implement would be approximately $1.5M or greater.

Based on the review by the SAMA Expert Panel, the costs to implement the modification are not 'minor'; therefore, the ER is revised to delete the statements that the costs to reactivate the automatic switchover feature would be minor.

See the Enclosure to this letter for the revision to the DBNPS LRA.

Question RAI 6.c

SAMA AC/DC-25, "provide a dedicated DC power system (battery/battery charger) for the TDAFW control valve and NNI-X for steam generator level indication," has an estimated implementation cost of $2M. This cost seems quite high for a system dedicated to just the TDAFW control valves and in light of the estimated costs for AC/DC-01 and AC/DC-03. Provide a more detailed description of this modification and justification for the estimated cost. Also, consider whether a portable system can provide the same benefit at a lower cost.

RESPONSE RAI 6.c

The Expert Panel made the following assumptions regarding SAMA candidate AC/DC-25 to provide a dedicated DC power system (battery/battery charger) for the TDAFW control valve and NNI-X for steam generator level indication:

- the DC power system will consist of a dedicated set of batteries and a battery charger;
- the intent of this SAMA would be to extend TDAFW pump operating time in the event of an SBO event, or loss of DC power to a TDAFW pump. Therefore, the dedicated DC system must have a longer battery lifetime than the existing safety-related DC system, or be able to supply power following loss of the current safety-related DC system;
- automatic steam generator level control will be needed (pump control, valves, indications, and speed changer motor, which means more DC power is required) to make the probabilistic risk assessment (PRA) case that the TDAFW pumps and level control are reliable;
- safety-related space for the batteries will be required (approximately $400K);
- major safety-related modification with seismic evaluation and calculation support required (approximately $500K);
- procedure changes with Engineering support and testing or training required (approximately $300K);
- batteries and other components and equipment, cable and conduit, disconnects to transfer DC power, including installation (approximately $700K); and
- both batteries / trains affected (additional costs).

Estimated cost to implement would be approximately $2M or greater.

A portable system, such as a diesel-driven battery charger or generator was evaluated in AC/DC-03, and was determined to cost approximately $330K or greater, and is considered cost-beneficial. For SAMA candidate AC/DC-25, due to the additional loads described above, an assumed portable system for this SAMA may require a larger generator unit to carry the loads. A portable system was not considered for this SAMA, however, because of the wording of the SAMA (i.e., a dedicated DC power system (battery/battery charger)).

Question RAI 6.d

SAMA CW-24 , "replace the standby CCW pump with a pump diverse from the other two CCW pumps," has an estimated implementation cost of $7.5M. This cost seems quite high for a pump replacement. Provide a more detailed description of this modification and justification for the estimated cost.

RESPONSE RAI 6.d

The Expert Panel made the following assumptions regarding SAMA candidate CW-24 to replace the standby CCW pump with a pump diverse from the other two CCW pumps:

- merely changing the standby pump with a different style pump would not meet the intent of the SAMA;
- additional safety-related space is needed that is separate from the existing component cooling water pumps due to the lack of space in the CCW pump room and to eliminate the potential for a common failure (i.e., flood) of all CCW pumps (approximately $2M);
- a new design pump, piping, valves and fittings will be required; cable and conduit required; components and equipment, including installation (approximately $4M);
- major safety-related modification with seismic evaluation and calculation support required (approximately $1M);
- procedure changes with Engineering support and testing or training required (approximately $500K);

Estimated cost to implement would be approximately $7.5M or greater.

Question RAI 6.e

As reported in Table E.7-2, the population dose risk reduction is either 10.00% (for 3 SAMAs) or 0.00% (for all other SAMAs). Explain how population dose risk was calculated and justify the result for each SAMA individually.

RESPONSE RAI 6.e

The results presented in Table E.7-2 appeared to be binary (either 0.00 percent or 10.00 percent). These population dose risk reduction values are correct, however, due to rounding in the Excel spreadsheet, the distinction between values for each SAMA candidate was not evident.

The population dose risk for each SAMA candidate is determined as follows:

1. The population dose is determined by execution of MACCS2 for each release category.
2. A PRA run for each SAMA candidate generates a new "vector" of release category frequencies.
3. The population dose risk (for each SAMA candidate) equals the sum (over all release categories) of the population dose for release category i times the frequency for release category i.

The percent change is determined by comparison of the population dose risk for each SAMA candidate compared with the base case (comparison dose). As the input from MACCS2 has changed (see response to RAI 4.b, above), the results presented in Table E.7-2 are revised; see the Enclosure to this letter for the revision to Table E.7-2. Note that the number of significant digits for the population dose (Off-site Annual Dose) provided in Table E.7-2 has increased to permit a discernable distinction between the population dose risk values for each SAMA candidate.

Question RAI 6.f

The model approach for SAMA AC/DC-01, "provide additional DC battery capacity," assumes a seven hour battery life. Provide the battery life assumed in the base PRA model, the basis for assuming a seven hour battery life in the SAMA analysis, and justification for the estimated implementation cost of $1.75M.

RESPONSE RAI 6.f

Davis-Besse has 4 Essential Batteries (1P, 1N, 2P & 2N). The four 125V DC, 1500 ampere-hour, lead-calcium batteries are provided and arranged to form two independent 125/250V DC Motor Control Centers (MCC). The batteries are sized to supply the anticipated DC and Instrument AC supply for a period of one hour after the loss of the battery charger supply. As discussed in FENOC procedure DB-OP-02521, "Loss of AC Bus Power Sources," non-essential loads can be shed to prolong battery life during a station blackout. The PRA assumes a 1 hour battery life. And, as discussed in USAR Chapter 15.2.9, decay heat removal after coastdown of the reactor coolant pumps is provided by natural circulation due to the raised loop design of Davis-Besse; the turbine-driven auxiliary feedwater pumps provide feedwater to the steam generators by taking suction from the condensate storage tanks. Feedwater level control can be provided by DC power, or manually. FENOC procedure DB-OP-02600, "Operational Contingency Response Action Plan," Attachment 1, "Emergency Control of Auxiliary Feedwater," identifies AFW System manual control actions, and Attachment 2, "Providing RPS/NNI Emergency Power Source," identifies actions to line up a portable

gasoline-powered AC generator (located in the Fire Brigade Equipment Room) to support manual operation of the AFW System following a loss of all AC and DC power.

A 6 - 8 hr battery was considered a reasonable extension for additional DC battery capacity based on the likelihood of recovering off-site power in this timeframe; SAMA AC/DC-01 considered 7 hrs.

The SAMA Expert Panel made the following assumptions regarding SAMA candidate AC/DC-01 to provide additional DC battery capacity:

- consider moving nonsafety-related loads to a new nonsafety-related battery;
- additional safety-related space for the batteries will be required; no space exists for additional batteries in the current battery room (approximately $500K);
- major modification required (approximately $200K);
- procedure changes with Engineering support and testing or training required (approximately $300K);
- batteries and other components and equipment, cable and conduit, including installation (approximately $600K); and,
- both batteries / trains affected – additional costs.

Estimated cost to implement would be approximately $1.75M or greater.

Question RAI 6.g

The model approach for SAMA AC/DC-14, "install a gas turbine generator," assumes failure of the station blackout (SBO) diesel generator is eliminated. This assumption does not provide credit for the gas turbine generator in the situation where all the emergency diesel generators (EDGs) are unavailable. Provide an assessment of the impact of this omission.

RESPONSE RAI 6.g

The Davis-Besse SBODG is manually started and loaded to supply power to Bus D2 in the event of an SBO. The SBODG is also available to power either shutdown Bus C1 or D1 at the onset of an SBO. In the Davis-Besse PRA, the SBODG is modeled as a backup to either EDG 1 or 2; it is considered in cases where both or either EDG 1 or 2 are unavailable . By eliminating failure of the SBODG (i.e., assuming it is perfectly reliable and available), this SAMA already accounts for crediting a gas turbine generator by ensuring one train of emergency power.

Question RAI 6.h

The model approach for SAMA CB-21, "install pressure measurements between the two DHR suction valves in the line from the RCS hot leg," assumes latent failures of the upstream valve are eliminated. It is unclear what is meant by "latent failures." Provide a more detailed description of the PRA model changes made to evaluate this SAMA.

RESPONSE RAI 6.h

The DHR ISLOCA model considers combinations of failures of the two motor-operated suction isolation valves in the DHR drop line. The valves are in series, so both must fail to result in an ISLOCA. Since both valves must fail, one valve could have failed at some point in the past without being detected as long as the other is not failed; this is what is meant by "latent failures." The failure of the other valve would then be the initiating event for the ISLOCA.

SAMA CB-21 proposed installing pressure indication in the piping between the two valves, which is not normally at RCS pressure. The pressure indication could detect if the inboard isolation valve (DH12) connected to the RCS had failed since startup, either by having failed to close while indicating closed, or by an internal rupture after startup. The analysis for SAMA CB-21 eliminated these failures of DH12, assuming that the failure would be detected and the unit shut down before the outboard isolation valve (DH11) fails. The PRA model also considers the case where DH12 fails, and the sudden increase in pressure on DH11 causes it to fail immediately. These failures were not removed from the cutsets because pressure indication would not serve to prevent the ISLOCA in that case.

Question RAI 6.i

i. **ER Section E.8.6 discusses six sensitivity cases. Relative to these sensitivity cases, provide the following:**

i. **Insufficient information is provided to understand the specific changes made to the baseline analysis assumptions for the first and fourth sensitivity cases. Provide a more detailed description of the analysis assumptions and methodology for these two cases.**

ii. **The description of the sixth sensitivity case states that off-site economic cost was increased by 25 percent. Table E.8-1 indicates that the total benefit for each of the SAMA candidates was increased by the same amount of $19,632, the offsite economic cost (AOC) value. Clarify how the**

increase of 25 percent in off-site economic cost correlates to the increase in total benefits of $19,632 for each SAMA.

RESPONSE RAI 6.i

6.i.i

The first sensitivity case in Section E.8.6 investigated the impact of assuming damaged plant equipment is repaired and refurbished following an accident scenario, as opposed to automatically decommissioning the facility following the event. For the purpose of this sensitivity case, the cost of repair and refurbishment over the lifetime of the plant is equivalent to 20 percent of the replacement power cost in accordance with NUREG/BR-0184. To calculate the benefit for the first sensitivity case, 20 percent of the replacement power cost from the baseline analysis for each SAMA candidate is used to estimate the repair and refurbishment costs.

The fourth sensitivity case in Section E.8.6 investigated the sensitivity of each analysis to the cost of replacement power. To determine the replacement power cost in 2009 dollars, the variable string power cost calculated in Section E.4.4.2 was modified for energy price inflation. The inflation rate was determined by assessing the electricity costs in 1993 and in 2009. The retail electricity cost for the state of Ohio in 1993 was 6.22 cents/kW-h and in 2009 was 8.96 cents/kW-h. The inflation rate was calculated using the method shown below:

$$z = \frac{2009\text{cost}}{1993\text{cost}} = \frac{8.96\text{cents/kW}-\text{h}}{6.22\text{cents/kW}-\text{h}} = 1.44$$

$$(1+x)^{(\Delta y)} = z$$

$$(1+x)^{(2009-1993)} = 1.44$$

$$x = 0.0231 \Rightarrow 2.31\%$$

$$y = \text{year}$$

$$x = \text{inflation rate}$$

The next step calculated the 2009 value for the string of replacement power costs based on the calculated inflation rate. The inflation of the string of replacement power costs (B) scaled for Davis-Besse was calculated using the equation shown below. The 2009 value for the string of the replacement power costs (B2009) was used to determine the present value of replacement power costs (PVRP) in 2009 dollars with a seven percent discount rate.

$$B_{2009} = B_{1993}(1+0.0231)^{(2009-1993)}$$

$$B_{2009} = (1.20E+08)(1+0.0231)^{(16)}$$

$$B_{2009} = \$1.73E+08$$

6.i.ii

The sixth sensitivity case investigated the sensitivity of the analysis to the off-site economic cost. For each SAMA candidate, a delta between the maximum benefit value and the specific SAMA candidate value is used to estimate the benefit for each SAMA candidate. This sensitivity case increased the maximum benefit off-site economic cost (AOC) value by 25 percent. When performing the delta calculation between the 25 percent increase to the maximum benefit AOC and AOC best-estimate value for each SAMA candidate, the total benefit increases by a constant value.

For example, for SAMA candidate AC/DC-01, the increased AOC value is $1,800 * 1.25 = $2,250. From this value, the AC/DC-01-specific off-site annual economic loss (property loss) value of $1,790 is subtracted, yielding a delta of $460. This value is compared to the base case delta calculation ($1,800 – $1,790 = $10). The total benefit increase when comparing the base case to the sensitivity case (for internal events) is $450 ($460 – $10 = $450); the total increase considering fire, seismic and other external events (multiplier of 4.6) is $450 + ($450 * 4.6) = $2,520. This value is then multiplied by the present worth factor of 12.27 to yield an increase of $30,920, as shown in Table E.8-1. Since the specific SAMA candidate off-site economic cost is included in both the base case calculation and the sensitivity case calculation, when subtracted, it yields a constant increase in the benefit for each SAMA candidate.

Since the cost-benefit analysis was revised with the results from the Level 3 PRA (see response to RAI 4.b), the constant value differs from the $19,632 stated in the RAI. The revised results are provided in the LRA mark-up of Table E.8-1 in the response to RAI 4.b.

Question RAI 6.j

ER Section 8.3 discusses a sensitivity case using a higher evacuation speed. Provide the evacuation speed used for this analysis. Also, Table E.3-31 shows that the population dose decreased compared to the base case yet Table E.8-1 shows the total net benefit increased by $1,963 for each SAMA. Explain this anomalous result and describe the methodology for developing the $1,963 used for each SAMA.

RESPONSE RAI 6.j

The evacuation speed used in the sensitivity case discussed in ER Section E.8.3 was 1.0 meter/second. The population dose used in the Section E.8.3 sensitivity case was the result of the Level 3 PRA sensitivity case E1.

As noted in the RAI, with a decrease in population dose, the net benefit for each SAMA candidate would be expected to decrease. The anomalous result (e.g., a net benefit increase) was due to the number of significant figures used in the Level 3 PRA and the cost-benefit evaluation. The population dose values differed in the third significant digit, which when rounded caused the unexpected results. As a result of the response to RAI 4.b, above, the population dose values have been revised for the Level 3 PRA sensitivity case E1. The ER revisions due to population dose were identified in the response to RAI 4.b.

With the revised results from RAI 4.b and consistent use of significant figures between the Level 3 PRA and cost-benefit analysis, the value $1963 is no longer germane to the sensitivity case in Section E.8.3.

As noted in the staff's RAI, a decrease in population dose was the result of sensitivity case E1 (where the evacuation speed was increased). Since NEI 05-01 suggested an evacuation speed sensitivity case to assess the impact on the results due to the uncertainty in the evacuation speed, it is logical to test (via a sensitivity case) the impact of a lower evacuation speed (which may cause a previously screened SAMA candidate to become cost-beneficial). Accordingly, the cost-benefit sensitivity case (Evacuation Speed from Table E.8-1) has been revised to use the results from Level 3 PRA sensitivity case E3, in which the evacuation speed is decreased by 9.6 percent, which causes a slight increase in population dose. ER Section E.3.5.2.4 is revised and new ER Table E.3-33 is added to incorporate sensitivity case E3.

The total benefit for each SAMA candidate has been increased by $1374, which is consistent with the increase in population dose. For the sensitivity case in Section E.8.3, the population doses values are taken from the Level 3 PRA sensitivity case E3 and replace the base case values in the determination of the averted public exposure (APE). Since there is a constant difference in the population dose values, for the Section E.8.3 sensitivity case, the total benefit for each SAMA is changed by the same

dollar amount. (See Table E.8-1 for results of evacuation speed sensitivity case in response to RAI 4.b.)

See the Enclosure to this letter for the revision to the DBNPS LRA.

Question RAI 6.k

The ER provides no assessment of the uncertainty distribution for CDF. Relative to the uncertainty distribution, address the following:

- **Provide the uncertainty distribution (5th, mean, and 95th percentiles) for the Davis-Besse PRA model CDF and describe how the distribution was developed.**

- **Provide an assessment of whether an uncertainty analysis using the 95th percentile CDF and the external events multiplier of 3.6 developed in RAI 3.c is bounded by the Multiplier Case sensitivity analysis. If not bounded, provide an uncertainty analysis using the 95th percentile CDF. In this analysis, provide an assessment of each Phase 1 SAMA eliminated using Screening Criterion D and E to determine whether any Phase 1 SAMAs originally screened should have a Phase 2 cost-benefit evaluation performed. Provide a Phase 2 cost-benefit evaluation for any SAMA not screened.**

- **If the Multiplier Case is bounding, provide an assessment of each Phase 1 SAMA eliminated using Screening Criteria D and E to determine whether any Phase 1 SAMAs originally screened should have a Phase 2 cost-benefit evaluation performed. Provide a Phase 2 cost-benefit evaluation for any SAMA not screened.**

RESPONSE RAI 6.k

The following table provides the uncertainty distribution for the Davis-Besse SAMA PRA model CDF. The 5th, mean, and 95th percentile values are in **bold** font:

	5% Conf.	Mean	95% Conf.
Point Estimate		9.70E-06	
Mean	1.06E-05	**1.07E-05**	1.09E-05
5th percentile	7.18E-06	**7.20E-06**	7.22E-06
Median	9.51E-06	9.53E-06	9.55E-06
95th percentile	1.53E-05	**1.55E-05**	1.56E-05
StdDev		1.48E-05	
Skewness		5.75E+01	
Kurtosis		4.55E+03	

The SAMA analysis model database was modified to support performance of an uncertainty analysis using the UNCERT software package. Failure rate distributions were entered into the database and modifications were made to make the database compatible with the UNCERT software. The SAMA analysis level 1 model was re-quantified to provide a cutset file compatible with the UNCERT software, and the uncertainty analysis was performed using the revised cutset file and database.

An assessment of the impact of the 95th percentile CDF uncertainty for internal events was performed for Davis-Besse. The uncertainty factor was derived from a ratio of the 95th percentile CDF uncertainty (1.55E-05/yr) to the point estimate CDF (1.07E-05/yr) for internal events. The uncertainty factor used in this analysis was 1.45. The analysis also used an external events multiplier of 4.6 (see the response to RAI 3.c for additional information on the development of the external events multiplier). Table 6.k-1, below, provides the cost-benefit results for the 95th percentile CDF uncertainty factor case. Also, the Multiplier Case was updated using an external events multiplier of seven (7). Table 6.k-2, below, provides the Multiplier Case cost-benefit results. The results of the 95th percentile CDF uncertainty and Multiplier Case sensitivity analyses identified one SAMA candidate (AC/DC-03) to be cost effective.

Since the external event multiplier used in the base case and the sensitivity case have changed, the issue of bounding is no longer relevant. Nonetheless, the SAMA candidates designated as Criterion D (Very Low Benefit) were re-evaluated (see Table 6.k-3, below) based on the results of the 95th percentile CDF uncertainty. For SAMA candidates where the 95th percentile CDF uncertainty basic event data were available, these basic events' RRW data were used as a basis for the final determination. For some SAMA candidates, either basic event data were not available, or basic event data were not applicable to the determination; for those cases, the determination basis is also provided.

SAMA candidate FL-01 (improve inspection of rubber expansion joints on main condenser) was initially identified for cost-benefit analysis based on the 95th percentile CDF uncertainty results. However, upon further investigation, the disposition of SAMA candidate FL-01 is changed to Criterion B (Already Implemented). The basis for the revised disposition is that the circulating water joints are currently inspected during outages and periodically replaced. ER Table E.6-1 is revised to include this change.

Further, based on additional information, SAMA candidate OT-05 (increase training and operating experience feedback to improve operator response) is changed from Criterion D (Very Low Benefit) to Criterion B (Already Implemented). The basis for the revised disposition is that Davis-Besse provides PRA information, such as risk significant initiating events, high worth operator actions and high worth equipment, to operators and other departments. Attachment 2 of FENOC procedure NOPM-CC-6000, "Probabilistic Risk Assessment Program," identifies items supported by the PRA Program; one item is PRA training support in areas such as new licensed operator training and operator re-qualification training cycles. ER Table E.6-1 is revised to include this change.

SAMA candidates screened with Criterion E (Subsumed) were addressed in the response to RAI 5.g, above.

See the Enclosure to this letter for the revision to the DBNPS LRA.

Table 6.k-1: 95th Percentile Uncertainty Factor Cost-Benefit Results

SAMA ID	95th Percentile Uncertainty Factor Estimated Benefit	Estimated Cost	Conclusion
AC/DC-01	$136,827	$1,750,000	Not Cost Effective
AC/DC-03	$548,194	$330,000	Cost Effective
AC/DC-14	$284,503	$2,000,000	Not Cost Effective
AC/DC-19	$48,930	$700,000	Not Cost Effective
AC/DC-21	$68,912	$100,000	Not Cost Effective
AC/DC-25	$341,569	$2,000,000	Not Cost Effective
AC/DC-26	$341,569	$2,000,000	Not Cost Effective
AC/DC-27	$0	$550,000	Not Cost Effective
CB-21	$46,827	$550,000	Not Cost Effective
CC-01	$2,989	$6,500,000	Not Cost Effective
CC-04	$0	$5,500,000	Not Cost Effective
CC-05	$0	$6,500,000	Not Cost Effective
CC-19	$21,974	$1,500,000	Not Cost Effective
HV-01	$0	$50,000	Not Cost Effective
HV-03	$0	$400,000	Not Cost Effective
AC/DC-28R	$393,488	$1,600,000	Not Cost Effective
CB-22R	$141,643	$4,600,000	Not Cost Effective
CC-22R	$0	$2,200,000	Not Cost Effective
CW-26R	$512,381	$1,500,000	Not Cost Effective
FW-17R	$584,227	$2,800,000	Not Cost Effective
OT-08R	$1,001,682	$4,400,000	Not Cost Effective

Table 6.k-2: Multiplier Case Cost-Benefit Results

SAMA ID	Multiplier Case	Estimated Cost	Conclusion
AC/DC-01	$134,805	$1,750,000	Not Cost Effective
AC/DC-03	$540,092	$330,000	Cost Effective
AC/DC-14	$280,299	$2,000,000	Not Cost Effective
AC/DC-19	$48,207	$700,000	Not Cost Effective
AC/DC-21	$67,893	$100,000	Not Cost Effective
AC/DC-25	$336,521	$2,000,000	Not Cost Effective
AC/DC-26	$336,521	$2,000,000	Not Cost Effective
AC/DC-27	$0	$550,000	Not Cost Effective
CB-21	$46,135	$550,000	Not Cost Effective
CC-01	$2,945	$6,500,000	Not Cost Effective
CC-04	$0	$5,500,000	Not Cost Effective
CC-05	$0	$6,500,000	Not Cost Effective
CC-19	$21,649	$1,500,000	Not Cost Effective
HV-01	$0	$50,000	Not Cost Effective
HV-03	$0	$400,000	Not Cost Effective
AC/DC-28R	$387,673	$1,600,000	Not Cost Effective
CB-22R	$139,550	$4,600,000	Not Cost Effective
CC-22R	$0	$2,200,000	Not Cost Effective
CW-26R	$504,809	$1,500,000	Not Cost Effective
FW-17R	$575,593	$2,800,000	Not Cost Effective
OT-08R	$986,879	$4,400,000	Not Cost Effective

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit"			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
Enhancements Related to AC and DC Power			
AC/DC-08	Increase training on response to loss of 120V AC buses that cause inadvertent actuation signals.	Criterion D Very Low Benefit	Abnormal Procedure DB-OP-2532 addresses the loss of both AC and DC power to both the Non-Nuclear Instrumentation (NNI) and the ICS that are powered from uninterruptible AC instrumentation distribution panels YAU and YBU. It is judged that operator awareness to the required actions is well established. *This SAMA should remain Criterion D.*
AC/DC-16	Improve uninterruptible power supplies.	Criterion D Very Low Benefit	Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to uninterruptible power supplies has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
Enhancements Related to ATWS Events			
AT-01	Add an independent boron injection system.	Criterion D Very Low Benefit	Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to emergency boration has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
AT-02	Add a system of relief valves to prevent equipment damage from pressure spikes during an ATWS.	Criterion D Very Low Benefit	Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to ATWS pressure relief has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
AT-07	Install motor generator set trip breakers in control room.	Criterion D Very Low Benefit	Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to reactor trip has an RRW value above the minimum cost of a hardware modification Also, if the reactor power is not decreasing, procedures instruct the operators to first de-energize substations E2 and F2, and, if necessary, locally open reactor trip breakers in the Low Voltage Switchgear room. *This SAMA should remain Criterion D.*
Enhancements Related to Containment Bypass			
CB-02	Add redundant and diverse limit switches to each CIV.	Criterion D Very Low Benefit	Failure of containment isolation typically leads to a LERF if core damage has occurred. LERF results are dominated by containment bypass events such as SGTR and ISLOCA events. Containment isolation is not shown to be a significant contributor to LERF in the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CB-03	Increase leak testing of valves in ISLOCA paths.	Criterion D Very Low Benefit	HPI and LPI injection check valves are leak tested per Appendix J. DHR suction lines are not tested, but rather than a leakage test, it is judged that continuously monitoring these valves at power would be preferable to leakage test. A SAMA candidate to continuously monitor the DHR suction valves is provided in SAMA candidate CB-21. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CB-04	Install self-actuating CIVs.	Criterion D Very Low Benefit	Important CIVs receive a close signal from the safety actuation system. Many are air-operated and fail in the closed position. It is judged that self-actuating valves would not provide any significant increase in the reliability of isolation. Containment isolation is not shown to be a significant contributor to CDF or LERF in the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CB-06	Ensure ISLOCA releases are scrubbed. One method is to plug drains in potential break areas so that break point will be covered with water.	Criterion D Very Low Benefit	This SAMA candidate would have very little benefit. It is likely that the break would be well above floor drain level. Therefore, a significant height of water would be required before any scrubbing took place. At these levels, the water level would likely have undesirable effects, such as threatening mitigating equipment due to flooding. This conclusion remains valid for the 95% CDF uncertainty results. *This SAMA should remain Criterion D.*
CB-09	Institute a maintenance practice to perform a 100% inspection of steam generator tubes during each refueling outage.	Criterion D Very Low Benefit	Davis-Besse is scheduled to replace the steam generators in 2013, which would result in inspecting new steam generator tubes. Therefore, this SAMA candidate is considered very low benefit for Davis-Besse. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CB-18	Direct steam generator flooding after a SGTR, prior to core damage.	Criterion D Very Low Benefit	Flooding the SG prior to core damage could impact efforts to mitigate the SGTR. For example, flooding may present a risk to the operation of the TDAFW pumps by risking steam generator overfill. *Disposition of this SAMA candidate is addressed in the response to RAI 5.i.*
CB-19	Vent MSSVs in containment.	Criterion D Very Low Benefit	This SAMA candidate would result in plant decay heat being deposited into primary containment, resulting in a harsh environment. The possible advantages for SGTR will be offset by the negative impacts for other events where secondary steam is deposited into containment with intact steam generators. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CB-20	Install relief valves in the CCW system.	Criterion D Very Low Benefit	Based on the top 100 cutsets and component basic event importance, ISLOCA in the CCW is not significant risk contributor at Davis-Besse. An ISLOCA occurring in the CCW system is not a risk contributor in the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
Enhancements Related to Core Cooling Systems			
CC-11	Modify procedures to throttle LPI pumps earlier in medium or large break LOCAs to maintain BWST inventory.	Criterion D Very Low Benefit	Davis-Besse operators are prohibited from throttling LPI pumps earlier in medium or large break LOCAs to maintain BWST inventory. If BWST flow was throttled down to reduce flowrate, the additional time gained is approximately 20 minutes, which, from a PRA perspective, is of low benefit for a LOCA condition. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CC-13	Upgrade the chemical and volume control system to mitigate small break LOCAs.	Criterion D Very Low Benefit	The make-up system can be used to provide make-up to the RCS in the event of a small LOCA. Because of the separate HPI and make-up systems, the plant has essentially four separate systems capable of injecting from the BWST into the RCS at high pressure. This was identified as a unique safety feature in the IPE. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CC-21	Reduce the BWST level at which switchover to containment recirculation is initiated.	Criterion D Very Low Benefit	Reducing the level at which switchover occurs (nine feet) would not significantly extend the time to switchover, and would increase the probability of pump failure due to loss of suction head. Davis-Besse has installed more accurate BWST level instrumentation that allows reaching a lower level prior to switchover to recirculation. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
Enhancements Related to Containment Phenomena			
CP-03	Use the fire water system as a backup source for the containment spray system.	Criterion D Very Low Benefit	Davis-Besse has a very large dry containment. Containment over-pressurization is not a significant risk contributor. This conclusion remains valid for the 95% LERF uncertainty case. *This SAMA should remain Criterion D.*
CP-06	Enhance fire protection system hardware and procedures.	Criterion D Very Low Benefit	This SAMA candidate addresses the scrubbing of radioactive releases into certain areas by actuating the fire protection system. Although some scrubbing benefits might be realized, this SAMA candidate presents the risk of impacting required equipment by spray or flooding. This could only be performed with fire protection systems that could be remotely actuated. If the temperature in certain areas became high enough, some existing fire protection systems may automatically actuate. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CP-16	Delay containment spray actuation after a large break LOCA.	Criterion D Very Low Benefit	The delay time that could be realized if containment spray was delayed would be less than 10 minutes. This SAMA candidate is considered to be of very low benefit. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CP-17	Install automatic containment spray pump header throttle valves.	Criterion D Very Low Benefit	The capability already exists at Davis-Besse to throttle containment spray after the switchover to the sump. The delay time that could be realized if containment spray was throttled would be less than 10 minutes. This SAMA candidate is considered to be of very low benefit. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CP-19	Install a redundant containment fan system.	Criterion D Very Low Benefit	Based on component basic event importance, containment fan coolers are not significant risk contributors at Davis-Besse. This SAMA candidate is considered to be very low benefit. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CP-20	Install or use an independent power supply to the hydrogen control system using either new batteries, a non-safety grade portable generator, existing station batteries, or existing AC/DC independent power supplies, such as the security system diesel generator.	Criterion D Very Low Benefit	Davis-Besse has a very large dry containment. Hydrogen burn does not present a significant risk in terms of LERF. This SAMA candidate is considered to be very low benefit. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CP-21	Install a passive hydrogen control system.	Criterion D Very Low Benefit	This SAMA would mitigate large early releases resulting from a hydrogen burn. LERF is dominated by containment bypass events such as SGTR and ISLOCA. Failure of containment is not a significant contributor to LERF. This SAMA candidate is considered to be very low benefit. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
Enhancements Related to Cooling Water			
CW-01	Add redundant DC control power for service water pumps.	Criterion D Very Low Benefit	Failure of DC power would impact much more than service water and improving the reliability of DC power to only service water would have very limited value. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to service water performance has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
CW-04	Add a redundant service water pump.	Criterion D Very Low Benefit	Davis-Besse has three service water pumps. In addition, the normally running cooling tower makeup pump is the preferred supply of service water following loss of service water. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to service water performance has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CW-05	Enhance the screen wash system.	Criterion D Very Low Benefit	The Davis-Besse water supply from Lake Erie travels through a long canal before reaching the intake structure. There is a screen at the intake from Lake Erie. The long distance traveled through the canal results in a significant fraction of material passing through the initial screen settling out prior to reaching the intake structure. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to service water performance has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
CW-06	Cap downstream piping of normally closed CCW drain and vent valves.	Criterion D Very Low Benefit	Loss of CCW through drain and vent lines is not considered to be a significant contributor to loss of CCW. These lines are small, and any leakage would likely be low. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CW-08	Enhance loss of CCW procedure to underscore the desirability of cooling down the RCS prior to seal LOCA.	Criterion D Very Low Benefit	Seal LOCA is not a concern at Davis-Besse if the RCPs are tripped. On loss of CCW, the makeup pumps can continue operation for at least one hour. Therefore, if operators trip the RCPs within one hour of loss of CCW, an RCP seal LOCA is not a risk concern. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CW-09	Additional training on loss of CCW.	Criterion D Very Low Benefit	Seal LOCA is not a concern at Davis-Besse if the RCPs are tripped. On loss of CCW, the makeup pumps can continue operation for at least one hour. Therefore, if operators trip the RCPs within one hour of loss of CCW, an RCP seal LOCA is not a risk concern. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
CW-12	Increase charging pump lube oil capacity.	Criterion D Very Low Benefit	Davis-Besse makeup pumps can operate for at least one hour on loss of CCW. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to charging (make-up) pump performance has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
CW-15	Use existing hydro test pump for RCP seal injection.	Criterion D Very Low Benefit	Seal LOCA is not a concern at Davis-Besse if the RCPs are tripped. On loss of CCW, the makeup pumps can continue operation for at least one hour. Therefore, if operators trip the RCPs within one hour of loss of CCW, an RCP seal LOCA is not a risk concern. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
CW-18	Prevent make-up pump flow diversion through the relief valves.	Criterion D Very Low Benefit	The make-up system is continuously operating. Malfunctions of relief valves would be immediately detected during operation and corrected. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related make-up flow diversion has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
Enhancements Related to Internal Flooding			
FL-01	Improve inspection of rubber expansion joints on main condenser.	*Revised to read:* Criterion F Considered for Further Evaluation	A large circulating water flood in the turbine building has associated basic event FL7 that is above the minimum cost of a procedure change (although less that a hardware modification). This SAMA candidate will be considered for further evaluation.
Enhancements Related to Fire Risk			
FR-01	Replace mercury switches in fire protection system.	Criterion D Very Low Benefit	Inadvertent actuation of fire protection water is not considered risk significant and currently not modeled in the PRA. Any fire protection system water should be handled by existing drains and is not considered a significant flooding threat. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
FR-02	Upgrade fire compartment barriers.	Criterion D Very Low Benefit	The Davis-Besse IPEEE did not identify any weakness in the fire barrier performance. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
FR-03	Install additional transfer and isolation switches.	Criterion D Very Low Benefit	Currently, isolation switches exist for a control evacuation. Some manual actions beyond operation of isolation switches are required (e.g., plugging connectors, removing/inserting fuse blocks). Adding additional transfer/isolation switches is not considered to be of significant benefit. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
FR-04	Enhance fire brigade awareness.	Criterion D Very Low Benefit	The Davis-Besse IPEEE did not identify any weakness in fire brigade performance. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
FR-05	Enhance control of combustibles and ignition sources.	Criterion D Very Low Benefit	The Davis-Besse IPEEE did not identify any weakness in the combustible control program. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
Enhancements Related to Feedwater and Condensate			
FW-03	Install an independent diesel for the CST make-up pumps.	Criterion D Very Low Benefit	Davis-Besse has the capability of replenishing the CST using fire protection water. This can be done even on loss of AC power. Adding diesel for condensate makeup pumps would add little benefit. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
FW-05	Install manual isolation valves around the TDAFW pump steam admission valves.	Criterion D Very Low Benefit	The purpose of the SAMA candidate was to reduce dual turbine-driven pump maintenance unavailability. Although manual isolation valves do not exist, Davis-Besse has valves within the steam lines that allow isolation of one TDAFW pump for maintenance while leaving the second TDAFW pump available. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
FW-07	Install a new condensate storage tank (AFW storage tank).	Criterion D Very Low Benefit	Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to CST performance has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
FW-12	Change failure position of condenser make-up valve if the condenser make-up valve fails open on loss of air or power.	Criterion D Very Low Benefit	On loss of air or electric power, several components required for secondary heat removal would be lost; therefore the state of the condenser make-up valve is not relevant. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
FW-15	Replace existing pilot-operated relief valves with larger ones, such that only one is required for successful feed and bleed.	Criterion D Very Low Benefit	Failure of the PORV to open only shows up in the Level 1 PRA importance measures with a RRW of 1.006 (cutoff 1.005). It does not show up in the top cutsets or the LERF importance list. Therefore, it is judged to be very low benefit. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to PORV opening or capacity has an RRW value above the minimum cost of a hardware modification *This SAMA should remain Criterion D.*
Enhancements Related to Heating, Ventilation and Air Conditioning (HVAC)			
HV-04	Add a switchgear room high temperature alarm.	Criterion D Very Low Benefit	The high voltage switchgear rooms do not require forced ventilation. Low voltage switchgear rooms require forced ventilation. Operators monitor the temperature of the low voltage switchgear rooms during their plant tours. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to switchgear ventilation has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
HV-05	Create ability to switch emergency feedwater room fan power supply to station batteries in an SBO.	Criterion D Very Low Benefit	Loss of ventilation to AFW is not a risk significant contributor at Davis-Besse. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
HV-06	Provide procedural guidance for establishing an alternate means of room ventilation to the service water pump room.	Criterion D Very Low Benefit	Service water ventilation includes four 50% fans. Loss of service water ventilation is not a significant risk contributor at Davis-Besse. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to service water room ventilation has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
Enhancements Related to Instrument Air and Nitrogen Supply			
IA-02	Modify procedure to provide ability to align diesel power to more air compressors.	Criterion D Very Low Benefit	Service Air and Instrument Air are not significant risk contributors based on top cutsets and risk importance measures. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to air compressors has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*
IA-03	Replace service and instrument air compressors with more reliable compressors that have self-contained air cooling by shaft-driven fans.	Criterion D Very Low Benefit	Service Air and Instrument Air are not significant risk contributors based on top cutsets and risk importance measures. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to service or instrument air compressors has an RRW value above the minimum cost of a hardware modification *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
Enhancements Related to Seismic Risk			
SR-01	Increase seismic ruggedness of plant components.	Criterion D Very Low Benefit	The Seismic Qualifications Utility Group (SQUG) previously identified the need for additional seismic restraints in the plant. These restraints have already been added. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
SR-02	Provide additional restraints for CO_2 tanks.	Criterion D Very Low Benefit	The CO_2 tanks are located outdoors. These tanks supply only the turbine generator. No other components are protected with CO_2. A seismic failure of the CO_2 tanks has minimal risk. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
Other Enhancements			
OT-01	Install digital large break LOCA protection system.	Criterion D Very Low Benefit	Large break LOCA is not a significant risk contributor (0.2% CDF). Davis-Besse has a Containment Leakage Detection System (FLUS) to identify leaks from vessel penetrations and nozzles. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*
OT-04	Improve maintenance procedures.	Criterion D Very Low Benefit	Davis-Besse has a qualified Maintenance Rule program in place. No deficiencies in maintenance practices have been identified. This conclusion remains valid for the 95% CDF uncertainty case. *This SAMA should remain Criterion D.*

Table 6.k-3: Re-evaluation of SAMA Candidates Screened as "Very Low Benefit" (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
OT-05	Increase training and operating experience feedback to improve operator response.	*Revised to read:* Criterion B Already Implemented	FENOC provides PRA information, such as risk-significant initiating events, high worth operator actions and high worth equipment, to various departments, including Operations Training, and presents this information on posters throughout the plant.
OT-07	Install secondary side guard pipes up to the MSIVs.	Criterion D Very Low Benefit	Steam line breaks are not a significant contributor to CDF or LERF based on top cutsets or basic event importance. The derived benefit would not justify the implementation cost required. Based on the basic event RRW results from the 95% CDF uncertainty case, no basic event related to main steam breaks has an RRW value above the minimum cost of a hardware modification. *This SAMA should remain Criterion D.*

Item 7

For certain SAMAs considered in the ER, there may be lower-cost alternatives that could achieve much of the risk reduction at a lower cost. In this regard, discuss whether any lower-cost alternatives to those Phase II SAMAs considered in the ER would be viable and potentially cost-beneficial. Evaluate the following SAMAs (previously found to be potentially cost-beneficial at other Babcock and Wilcox plants), or indicate if the particular SAMA has already been considered. If the latter, indicate whether the SAMA has been implemented or has been determined to not be cost-beneficial at Davis-Besse Nuclear Power Station.

Question RAI 7.a

Automate reactor coolant pump trip on high motor bearing cooling temperature.

RESPONSE RAI 7.a

A SAMA candidate (CW-26R) to provide an automatic reactor coolant pump trip on loss of cooling to the RCP seal thermal barrier cooler and loss of seal injection flow was evaluated for Davis-Besse. Table 7.a-1 and Table 7.a-2, below, provide the internal event and total benefit results for SAMA candidate CW-26R, respectively. Table 7.a-3, below, provides the final results for the ten sensitivity cases for SAMA candidate CW-26R. The implementation cost for this SAMA candidate was estimated as $1,500,000. Therefore, this SAMA candidate is not cost-beneficial at Davis-Besse.

Table 7.a-1: Internal Events Benefit Results for SAMA Candidate CW-26R

Case	**CW-26R** (Auto_RCP)
Off-site Annual Dose (rem)	2.27E+00
Off-site Annual Property Loss ($)	1.79E+03
Comparison CDF	1.0E-05
Comparison Dose (rem)	2.30E+00
Comparison Cost ($)	1.80E+03
Enhanced CDF	7.7E-06
Reduction in CDF	**23.00%**
Reduction in Off-site Dose	**1.30%**
Immediate Dose Savings (On-site)	$186
Long Term Dose Savings (On-site)	$812
Total Accident Related Occupational Exposure **(AOE)**	$998
Cleanup/Decontamination Savings (On-site)	$30,443
Replacement Power Savings (On-site)	$30,801
Averted Costs of On-site Property Damage **(AOSC)**	$61,244
Total On-site Benefit	**$62,242**
Averted Public Exposure **(APE)**	$736
Averted Off-site Damage Savings **(AOC)**	$123
Total Off-site Benefit	**$859**
Total Benefit (On-site + Off-site)	**$63,101**

Table 7.a-2: Total Benefit Result for SAMA Candidate CW-26R

	CW-26R (Auto_RCP)
Internal Events	$63,101
Fires, Seismic, Other	$290,265
Total Benefit	$353,366

Table 7.a-3: Final Results of the Sensitivity Cases for SAMA Candidate CW-26R

SAMA ID	Repair Case	Low Discount Rate Case	High Discount Rate Case	On-site Dose Case	On-site Clean-up Case
CW-26R	$215,378	$533,291	$242,495	$359,021	$410,194

SAMA ID	Replacement Power Case	Multiplier Case	Evacuation Speed	Off-site Economic Cost	95th CDF Case
CW-26R	$469,142	$504,809	$354,741	$384,287	$512,381

Question RAI 7.b

Use the decay heat removal (DHR) system as an alternate suction source for high pressure injection (HPI).

RESPONSE RAI 7.b

The Davis-Besse design and PRA already include use of the DHR system as a suction source for HPI. For cases in which RCS pressure is too high for adequate flow, the HPI pumps can be aligned to take suction from the discharge of the DHR pumps; this is possible with the BWST as the suction source or with the containment sump as the suction source.

Question RAI 7.c

Automate HPI injection on low pressurizer level (in loss of secondary side heat removal cases where the reactor coolant system (RCS) pressure remains high while the RCS level drops) -Three Mile Island SAMA 16.

RESPONSE RAI 7.c

This SAMA candidate considers automating HPI injection on low pressurizer level following a loss of secondary side heat removal where RCS pressure remains high while level drops. This SAMA was a viable consideration for Three Mile Island (TMI)

based on plant design and system configuration. At TMI, the HPI system is also the makeup system – there is a single Makeup and Purification system that provides normal makeup as well as standby Engineered Safety Actuation Signal (ESAS)-selected pumps which automatically inject high-pressure water into the RCS from the BWST in mitigation of LOCA scenarios. In addition, as discussed in Volume 3 of the B&W Emergency Operating Procedure Technical Basis Document (EOP TBD), (Chapter III.C, Lack of Adequate Primary to Secondary Heat Transfer), for all plants except Davis-Besse, HPI cooling must not be intentionally delayed if feedwater is not available. HPI cooling must be established in a timely manner to assure adequate core cooling; it must be started early enough to slow RCS inventory depletion so that HPI cooling will match decay heat before the core is uncovered.

At Davis-Besse, however, the plant design and systems are different from those at TMI. Davis-Besse has a separate HPI safety system in addition to the normally operating makeup system. The Davis-Besse HPI system is not capable of injecting water into the RCS until pressure reaches ~1600psig. In addition, because Davis-Besse has two makeup pumps, makeup/HPI cooling can be delayed until the core outlet temperature reaches 600°F provided the RCS PT limit is not exceeded. Although the Davis-Besse PRA considers makeup/HPI cooling in response to a loss of feedwater, and the associated operator actions, automating this function was not considered because of the complexity associated with the number of options and systems involved (e.g., pumps, valves and alignment options, injection line options, bleed options). Consequently, this SAMA candidate was not considered for Davis-Besse.

Question RAI 7.d

Automate refill of the borated water storage tank (BWST).

RESPONSE RAI 7.d

A SAMA candidate (CC-22R) to provide an automatic refill of the borated water storage tank was evaluated for Davis-Besse. Table 7.d-1 and Table 7.d-2, below, provide the internal event and total benefit results for SAMA candidate CC-22R, respectively. Table 7.d-3, below, provides the final results for the ten sensitivity cases for SAMA candidate CC-22R. The implementation cost for this SAMA candidate was estimated as $2,200,000. Therefore, this SAMA candidate is not cost-beneficial at Davis-Besse.

Table 7.d-1: Internal Events Benefit Results for SAMA Candidate CC-22R

Case	CC-22R (Auto_BWST)
Off-site Annual Dose (rem)	2.30E+00
Off-site Annual Property Loss ($)	1.80E+03
Comparison CDF	1.0E-05
Comparison Dose (rem)	2.30E+00
Comparison Cost ($)	1.80E+03
Enhanced CDF	1.0E-05
Reduction in CDF	**0.00%**
Reduction in Off-site Dose	**0.00%**
Immediate Dose Savings (On-site)	$0
Long Term Dose Savings (On-site)	$0
Total Accident Related Occupational Exposure **(AOE)**	$0
Cleanup/Decontamination Savings (On-site)	$0
Replacement Power Savings (On-site)	$0
Averted Costs of On-site Property Damage **(AOSC)**	$0
Total On-site Benefit	**$0**
Averted Public Exposure **(APE)**	$0
Averted Off-site Damage Savings **(AOC)**	$0
Total Off-site Benefit	**$0**
Total Benefit (On-site + Off-site)	**$0**

Table 7.d-2: Total Benefit Result for SAMA Candidate CC-22R

	CC-22R (Auto_BWST)
Internal Events	$0
Fires, Seismic, Other	$0
Total Benefit	$0

Table 7.d-3: Final Results of the Sensitivity Cases for SAMA Candidate CC-22R

SAMA ID	Repair Case	Low Discount Rate Case	High Discount Rate Case	On-site Dose Case	On-site Clean-up Case
CC-22R	$0	$0	$0	$0	$0

SAMA ID	Replacement Power Case	Multiplier Case	Evacuation Speed	Off-site Economic Cost	95th CDF Case
CC-22R	$0	$0	$1,374	$30,920	$0

Question RAI 7.e

Automate start of auxiliary feedwater (AFW) pump in the event the automated emergency feedwater (EFW) system is unavailable.

RESPONSE RAI 7.e

A SAMA candidate (FW-17R) to automatically start the auxiliary feedwater pump when the emergency feedwater system is unavailable was evaluated for Davis-Besse. Based on the Davis-Besse design, this SAMA was interpreted as automatically starting the motor driven feedwater pump in the event both turbine-driven auxiliary feedwater pumps were not available. Table 7.e-1 and Table 7.e-2, below, provide the internal event and total benefit results for SAMA candidate FW-17R, respectively. Table 7.e-3, below, provides the final results for the ten sensitivity cases for SAMA candidate FW-17R. The implementation cost for this SAMA candidate was estimated as $2,800,000. Therefore, this SAMA candidate is not cost-beneficial at Davis-Besse.

Table 7.e-1: Internal Events Benefit Results for SAMA Candidate FW-17R

Case	FW-17R (Auto_MDFP)
Off-site Annual Dose (rem)	2.18E+00
Off-site Annual Property Loss ($)	1.69E+03
Comparison CDF	1.0E-05
Comparison Dose (rem)	2.30E+00
Comparison Cost ($)	1.80E+03
Enhanced CDF	7.5E-06
Reduction in CDF	**25.00%**
Reduction in Off-site Dose	**5.22%**
Immediate Dose Savings (On-site)	$202
Long Term Dose Savings (On-site)	$882
Total Accident Related Occupational Exposure **(AOE)**	$1,085
Cleanup/Decontamination Savings (On-site)	$33,091
Replacement Power Savings (On-site)	$33,479
Averted Costs of On-site Property Damage **(AOSC)**	$66,570
Total On-site Benefit	**$67,655**
Averted Public Exposure **(APE)**	$2,945
Averted Off-site Damage Savings **(AOC)**	$1,350
Total Off-site Benefit	**$4,294**
Total Benefit (On-site + Off-site)	**$71,949**

Table 7.e-2: Total Benefit Result for SAMA Candidate FW-17R

	FW-17R (Auto_MDFP)
Internal Events	$71,949
Fires, Seismic, Other	$330,966
Total Benefit	$402,915

Table 7.e-3: Final Results of the Sensitivity Cases for SAMA Candidate FW-17R

SAMA ID	Repair Case	Low Discount Rate Case	High Discount Rate Case	On-site Dose Case	On-site Clean-up Case
FW-17R	$252,928	$608,721	$277,988	$409,062	$464,684

SAMA ID	Replacement Power Case	Multiplier Case	Evacuation Speed	Off-site Economic Cost	95th CDF Case
FW-17R	$528,758	$575,593	$404,289	$433,835	$584,227

Question RAI 7.f

Purchase or manufacture of a "gagging device" that could be used to close a stuck-open steam generator safety valve for a SGTR event prior to core damage.

RESPONSE RAI 7.f

A SAMA candidate (CB-22R) to use a "gagging" device that could be used to close a stuck-open steam generator safety valve for a SGTR was evaluated for Davis-Besse. Table 7.f-1 and Table 7.f-2, below, provide the internal event and total benefit results for SAMA candidate CB-22R, respectively. Table 7.f-3, below, provides the final results for the ten sensitivity cases for SAMA candidate CB-22R. The implementation cost for this SAMA candidate was estimated as $4,600,000. The high implementation cost of this SAMA candidate is based on replacement of the safety valves with a new design that includes a gagging feature. Therefore, this SAMA candidate is not cost-beneficial at Davis-Besse.

Table 7.f-1: Internal Events Benefit Results for SAMA Candidate CB-22R

Case	CB-22R (Gagging_Device)
Off-site Annual Dose (rem)	2.04E+00
Off-site Annual Property Loss ($)	1.56E+03
Comparison CDF	1.0E-05
Comparison Dose (rem)	2.30E+00
Comparison Cost ($)	1.80E+03
Enhanced CDF	9.7E-06
Reduction in CDF	**3.00%**
Reduction in Off-site Dose	**11.30%**
Immediate Dose Savings (On-site)	$24
Long Term Dose Savings (On-site)	$106
Total Accident Related Occupational Exposure **(AOE)**	$130
Cleanup/Decontamination Savings (On-site)	$3,971
Replacement Power Savings (On-site)	$4,018
Averted Costs of On-site Property Damage **(AOSC)**	$7,988
Total On-site Benefit	**$8,119**
Averted Public Exposure **(APE)**	$6,380
Averted Off-site Damage Savings **(AOC)**	$2,945
Total Off-site Benefit	**$9,325**
Total Benefit (On-site + Off-site)	**$17,444**

Table 7.f-2: Total Benefit Result for SAMA Candidate CB-22R

	CB-22R (Gagging_Device)
Internal Events	$17,444
Fires, Seismic, Other	$80,241
Total Benefit	$97,685

Table 7.f-3: Final Results of the Sensitivity Cases for SAMA Candidate CB-22R

SAMA ID	Repair Case	Low Discount Rate Case	High Discount Rate Case	On-site Dose Case	On-site Clean-up Case
CB-22R	$79,687	$149,212	$71,121	$98,423	$105,097

SAMA ID	Replacement Power Case	Multiplier Case	Evacuation Speed	Off-site Economic Cost	95th CDF Case
CB-22R	$112,786	$139,550	$99,059	$128,605	$141,643

Enclosure

Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS)

Letter L-11-154

Amendment No. 10 to the
DBNPS License Renewal Application

License Renewal Application
Environmental Report (ER) Sections Affected

Environmental Report	Section E.4.2	Table E.3-29
Section 4.20	Section E.4.5	Table E.3-30
Table 6.1-1	Section E.5.6	Table E.3-31
	Section E.9	Table E.3-32
ER Attachment D		Table E.3-33
Section D.2.1	**Section E.10**	Table E.4-1
	Table E.3-11	Table E.5-3
ER Attachment E	Table E.3-21	Table E.5-4
Executive Summary	Table E.3-22	Table E.6-1
Section E.3.1.1.1	Table E.3-23	Table E.7-2
Section E.3.1.2.4	Table E.3-24	Table E.7-3
Section E.3.2.1	Table E.3-25	Table E.7-5
Section E.3.4.2	Table E.3-26	Table E.8-1
Section E.3.5.2.4	Table E.3-27	
Section E.4.1	Table E.3-28	Section E.11

The Enclosure identifies the change to the License Renewal Application (LRA) by Affected LRA Section, LRA Page No., and Affected Paragraph and Sentence. The count for the affected paragraph, sentence, bullet, etc. starts at the beginning of the affected Section or at the top of the affected page, as appropriate. Below each section the reason for the change is identified, and the sentence affected is printed in *italics* with deleted text ~~*lined-out*~~ and added text *underlined*.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section 4.20	**4.20-3 & 4.20-4**	**Final paragraph**

In response to RAIs 4.b and 5.c, Environmental Report (ER) Section 4.20, "Severe Accident Mitigation Alternatives," final paragraph, is replaced in its entirety, and now reads:

The results of the evaluation of 168 SAMA candidates identified one cost-beneficial enhancement at Davis Besse. Assuming a lower discount rate, higher dose rates, higher onsite clean-up cost, increased replacement power costs, increased external event multiplier, increased off-site economic impact, and reduced evacuation speed identified the same SAMA candidate to be cost-beneficial. The SAMA candidate identified in the base case and sensitivity cases is not related to plant aging. Therefore, the identified cost-beneficial SAMA candidate is not a required modification for the license renewal period. Nevertheless, this SAMA candidate will be considered through the normal FENOC processes for evaluating possible modifications to the plant.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table 6.1-1	**6.1-5**	**Row 76, Environmental Impact column**

In response to RAI 4.b, ER Table 6.1-1, "Environmental Impacts Related to License Renewal at Davis-Besse," Row 76, Environmental Impact column, is revised to read:

No.	Category 2 Issue	Environmental Impact
Postulated Accidents		
76	Severe accident mitigation alternatives **10 CFR 51.53(c)(3)(ii)(L)**	**SMALL**. No impact from continued operation. FENOC ~~*did not identify any*~~ *identified one* cost-beneficial enhancement~~*s, but did identify one potential cost-beneficial SAMA candidate*~~, which FENOC will consider through normal processes for evaluating possible changes to the plant.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section D.2.1	**D-10**	**4th bullet on page**

In response to RAI 4.b, ER Section D.2.1, "Environmental Impacts – Background Information," last bullet in the Section, is revised to read:

- Severe accidents – The NRC determined that the license renewal impacts from severe accidents would be small, but that applicants should perform site-specific analyses of ways to further mitigate impacts. *Results from the FENOC severe accident mitigation alternatives (SAMA) analysis ~~have not~~ identified ~~any~~ one cost-beneficial enhancement~~s to~~ that may further mitigate risk to public health and the economy in the area of the plant, including the coastal zone, due to potential severe accidents at Davis Besse.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Attachment E – Executive Summary	**E-9**	**4th and 5th paragraphs**

In response to RAIs 4.b and 5.c, the Executive Summary of ER Attachment E, "Severe Accident Mitigation Alternatives Analysis," paragraphs four and five, are revised to read:

The cost-benefit evaluation of SAMA candidates performed for Davis-Besse provides significant insight into the continued operation of Davis-Besse. *The results of the evaluation of ~~167~~ 168 SAMA candidates indicate ~~no enhancements~~ one enhancement to be cost-beneficial for implementation at Davis-Besse. The cost-beneficial SAMA candidate is AC/DC-03, which adds a portable diesel-driven battery charger to the DC system.*

~~However, the~~ The sensitivity cases performed for this analysis found ~~one~~ the same SAMA candidate (AC/DC-03) to be cost-beneficial for implementation at Davis-Besse under the assumptions of ~~three of the sensitivity cases (lower discount rate, replacement power, and multiplier). SAMA candidate AC/DC-03 considered the addition of a portable diesel-driven battery charger for the DC~~

~~*system.*~~ *lower discount rate, higher dose rates, higher onsite clean-up cost, increased replacement power costs, increased external event multiplier, increased off-site economic impact, and reduced evacuation speed sensitivity cases.* While the identified SAMA candidate is not related to plant aging and therefore not required to be resolved as part of the relicensing effort, FENOC will, nonetheless, consider implementation of this candidate through normal processes for evaluating possible changes to the plant.

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
ER Section E.3.1.1.1	**E-19**	**Second paragraph, first sentence**

In response to RAI 1.e, ER Section E.3.1.1.1, "Description of Level 1 Internal Events PRA Model," second paragraph, first sentence, is replaced in its entirety, and now reads:

The Davis Besse Level 1 PRA internal events CDF, including internal flooding, is estimated to be 9.2E-06/yr, and when also including high winds, the CDF is estimated to be 9.8E-06/yr.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.3.1.2.4	**E-28**	**Entire section**

In response to RAI 3.c, ER Section E.3.1.2.4, "External Event Severe Accident Risk," is deleted in its entirety, as follows:

~~*E.3.1.2.4 External Event Severe Accident Risk*~~

~~*This section describes the method used to address external events risk.*~~

~~*As discussed in Section E.3.1.2.2, Davis Besse used the SMA to evaluate the risk from seismic events. While this methodology does not provide a quantitative result, the resolution of outliers ensures that the seismic risk is low and further cost-beneficial seismic improvements are not expected. Also, as discussed in Section E.3.1.2.3, no other external events were found to exceed the screening criteria. Therefore, the FIVE results were used as a measure of total external events risk.*~~

~~*As discussed in Section E.3.1.2.1, using the EPRI FIVE methodology, Davis Besse conservatively estimated the Fire CDF to be 2.5E-05/yr. Since the FIVE methodology contains numerous conservatisms, a more realistic assessment could result in a substantially lower fire CDF. As noted in NEI 05-01 (Reference 2), the NRC staff has accepted that a more realistic fire CDF may be a factor of three less than the screening value obtained from a FIVE analysis.*~~

~~*Based on the Davis Besse FIVE CDF of 2.5E-05/yr, a factor of three reduction would result in a fire CDF of approximately 8.3E-06/yr. This value is the same order of magnitude as the internal events CDF of 9.2E-06/yr. Therefore, this justifies use of an external events multiplier of three to the averted cost estimates (for internal events) to represent the additional SAMA benefits in external events.*~~

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.3.2.1	**E-30**	**Last paragraph**

In response to RAI 2.a, ER Section E.3.2.1, "Description of the Level 2 PRA Model," the last paragraph of the Section on page E-30, is revised to read:

The SAMA analysis model calculated a LERF of 6.6E-07/year. Table E.3-8 ranks the top 30 components for Level 2 PRA based on Fussell-Vesely importance measure. Table E.3-9 provides the top ten operator actions for Level 2 PRA ranked by Fussell-Vesely importance measure. *LERF was quantified using a truncation cutoff frequency of 5.0E-13/yr.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.3.4.2	**E-34**	**1st paragraph**

In response to RAI 4.b, ER Section E.3.4.2, "Population Data," first paragraph, is revised to read:

The population data were extracted using SECPOP2000 (Reference 18) with 2000 census data for Davis Besse sited at latitude of 41 degrees, 35 minutes, 50 seconds, and longitude of 83 degrees, 5 minutes, 11 seconds. *To the SECPOP2000 population, Canadian population data in sectors 30-40 miles/N, 30-40 miles/NNE, 30-40 miles/NE, 40-50 miles/N, 40-50 miles/NNE, and 40-50 miles/NE were added. The Canadian population was estimated by subtracting the SECPOP2000 population data from the total population in the 50-mile radius of Davis-Besse, as reported in Environmental Report Table 2.6-1. Population was assigned to each of the affected six sectors normalized by the land fraction in each of the sectors.* The population data were adjusted to account for the transient population within 10 miles of Davis Besse. The transient population segment, includes seasonal residents, transient population, and boating population. The population escalation factor was developed considering different sets of population data, e.g., state-wide versus within a 50-mile radius of the plant.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.3.5.2.4	**E-45**	**New paragraph**

In response to RAI 4.b, ER Section E.3.5.2.4, "Early," a new paragraph for sensitivity case E3 is added to the end of the section, which reads:

Case E3 *– The base case was performed with an evacuation speed of 0.58 meters/second, based on Davis-Besse-specific evaluation information, without any correction factor to account for the escalated population. In response to an NRC request for additional information, this sensitivity case was performed to gauge the sensitivity of reducing the evacuation speed. As the population was increased 4.7 percent per decade for the 20 years of license renewal (total increase of 9.6 percent), it was assumed for this sensitivity case that the increase in population was directly proportional to the decrease in evacuation speed. The evacuation speed for this sensitivity is a 9.6 percent decrease from the base case, i.e., 0.52 meters/second. This change resulted in a minor increase in the consequence values, as shown in Table E.3-33. This is expected as slower evacuation should remove the population from the radiological damage less quickly.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.4.1	**E-47**	**1st paragraph on page**

In response to RAI 4.b, ER Section E.4.1, "Off-site Exposure Cost," the first paragraph on page E-47, is revised to read:

Table E.3-21 provides the off-site dose for each release category obtained for the base case of the Davis Besse Level 3 PRA weighted by the release category frequency. *The total off-site dose for internal events (Dt) was estimated to be ~~2.0~~ 2.30 person-rem/year.* The APE cost was determined using Equation E.4-2 (Reference 1, Section 5.7.1).

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.4.1	**E-48**	**Equations E.4-6 and E.4-7**

In response to RAI 4.b, ER Section E.4.1, "Off-site Exposure Cost," equations E.4-6 and E.4-7, are replaced in their entirety, and now read:

$$Z_{pha} = \left(2{,}000 \frac{\$}{person-rem}\right)\left(2.30 \frac{person-rem}{yr}\right) = \$4600/yr \qquad (E.4\text{-}6)$$

where,

R = $2,000/person-rem

Dt = 2.30 person-rem/year

The values for the base case are:

C = 12.27 yr

Zpha = $4,600/yr

$$APE = (12.27yr)\left(\frac{\$4600}{yr}\right) = \$56{,}442 \qquad (E.4\text{-}7)$$

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.4.2	**E-49**	**1st paragraph and equations E.4-8 and E.4-9**

In response to RAI 4.b, ER Section E.4.2, "Off-site Economic Cost," the first paragraph and equations E.4-8 and E.4-9, are revised to read:

The term used for off-site economic cost is designated as averted off-site property damage costs (AOCs). The off-site economic loss for a 50-mile radius of the site was determined using the MACCS2 model developed for the Davis Besse Level 3 PRA in Section E.3.4. Table E.3-21 provides the economic loss for each release category obtained for the base case of the Level 3 PRA weighted by the release category frequency. *The total economic loss from internal events (It) was estimated to be ~~$1,600~~ $1,800 per year.* The averted cost was determined using Equation E.4-8 from Reference (1), Section 5.7.5.

$$AOC = (C)(I_t) \qquad \text{(E.4-8)}$$

where,

AOC = off-site economic costs associated with a severe accident ($)

C = present value factor (yr)

I_t = monetary value of economic loss per year from internal events before discounting ($/yr)

The values for the base case are:

C = 12.27 yr

I_t = ~~*$1,600/yr*~~ *$1,800/yr*

$$AOC = (12.27\text{yr})\left(1800\frac{\$}{\text{yr}}\right) = \$22{,}086 \qquad \text{(E.4-9)}$$

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.4.5	**E-55**	**Entire section, including equations**

In response to RAIs 3.c and 4.b, ER Section E.4.5, "Total Cost of Severe Accident Risk," is revised to read:

The total cost of severe accident impact for internal events was calculated by summing the public exposure cost, off-site property damage cost, occupational exposure cost, and on-site economic cost. *The cost of the impact of a severe accident for internal events was ~~$339,331~~ $349,147 as shown in Table E.4-1. Davis Besse does not have external events (fire, seismic, other external events) PRA from which risk contributors could be combined with the internal events risk. ~~This analysis assumed that the benefit from each hazard group's (i.e., fire, seismic, and other external events) contribution is equivalent to that of internal events. This approach is conservative, based on the discussion in Section E.3.1.2. Therefore, the cost of SAMA candidate implementation was compared with a benefit value of four times (i.e., 1x for internal events plus 3x for external events) that calculated for internal events to include the contribution from internal events, fire, seismic, and other hazard groups.~~ Based on the NRC staff's best estimate, the fire CDF for Davis-Besse is 2.9x10-5/yr [39]. To account for the risk contribution from the fire hazard, a ratio between the fire CDF and internal events CDF was used to determine a fire multiplier of 2.90 (see equation E.4-24).*

$$\frac{\text{Fire CDF}}{\text{Internal Events CDF}} = \frac{2.9\text{x}10^{-5}/\text{yr}}{1.0\text{x}10^{-5}/\text{yr}} = 2.90 \qquad \textit{(E.4-24)}$$

Based on updated probabilistic seismic hazard estimates due to Generic Issue 199, the NRC staff estimated a "weakest link model" seismic CDF for Davis-Besse of $6.7\text{x}10^{-6}$/yr [40]. To account for the risk contribution from the seismic hazard, a ratio between the seismic CDF and internal events CDF was used to determine a seismic multiplier of 0.67 (see equation E.4-25).

$$\frac{\text{Seismic CDF}}{\text{Internal Events CDF}} = \frac{6.7\text{x}10^{-6}/\text{yr}}{1.0\text{x}10^{-5}/\text{yr}} = 0.67 \qquad \textit{(E.4-25)}$$

This analysis conservatively assumed that the benefit from other hazard groups contribution is equivalent to that of internal events. Therefore, the other hazard groups multiplier is 1.0.

To determine the multiplier to account for fire, seismic, and other hazard groups, the individual multipliers are summed; the resulting multiplier is 4.6.

This approach provided a comparison of the cost to the risk reduction estimated for internal and external events for each SAMA candidate. *The maximum benefit for Davis Besse was ~~$1,357,324~~ $1,955,223 as shown in Table E.4-1.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.5.6	**E-63**	**1st sentence**

In response to RAIs 4.b and 5.c, ER Section E.5.6, "Initial SAMA Candidate List," the first sentence in the section is revised to read:

Based on the review of the aforementioned sources, an initial list of ~~167~~ 168 SAMA candidates was assembled.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.9	**E-74**	**1st and 2nd paragraphs**

In response to RAIs 4.b and 5.c, the first and second paragraphs of ER Section E.9, "Conclusions," are revised to read:

The cost-benefit evaluation of SAMA candidates performed for the Davis-Besse license renewal process provided significant insight into the continued operation of Davis-Besse. *The results of the evaluation of ~~167~~ 168 SAMA candidates indicated ~~no enhancements to be potentially~~ one enhancement to be cost-beneficial for implementation at Davis-Besse. The cost-beneficial SAMA candidate is AC/DC-03, which adds a portable diesel-driven battery charger to the DC system.*

~~However, the~~ The sensitivity cases performed for this analysis found ~~one~~ the same SAMA candidate (AC/DC-03) to be ~~potentially~~ cost-beneficial for implementation at Davis-Besse under the assumptions of the second (lower discount rate), fourth (higher discount rate), fifth (higher on-site clean-up cost), sixth (increased replacement power costs), seventh (increased external event multiplier), eighth (increased off-site economic impact), and ninth (reduced evacuation speed) sensitivity cases. ~~three of the sensitivity cases (low discount rate, replacement power, and multiplier). SAMA candidate AC/DC-03 considered the addition of a portable diesel-driven battery charger for the DC system.~~ While the identified SAMA candidate is not related to plant aging and therefore not a required modification for the license renewal period, FENOC will, nonetheless, consider implementation of this candidate through the normal processes for evaluating possible plant modifications.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Section E.11	**E-194**	**New references**

In response to RAI 3.c, ER Section E.11, "References," is revised to include two new references cited in revised ER Section E.4.5, as follows:

39. Nuclear Regulatory Commission, "Requests for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit 1, License Renewal Application," Accession Number ML110910566, April 20, 2011.

40. Nuclear Regulatory Commission, Results of Safety/Risk Assessment of Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," Accession Number ML100270582, September 7, 2010.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.3-11	**E-86**	**3 rows**

In response to RAI 4.b, three rows (i.e., N, NNE, and NE) in ER Table E.3-11, "Total (Permanent and Transient) Escalated Population (50-Mile Radius – Davis Besse) for the Year 2040," are revised to include the Canadian population within the Davis-Besse 50-mile Emergency Planning Zone, and now reads:

Table E.3-11: Total (Permanent and Transient) Escalated Population (50-Mile Radius – Davis-Besse) for the Year 2040

Sector	1 mile	2 miles	3 miles	4 miles	5 miles	10 miles	20 miles	30 miles	40 miles	50 miles
N	0	0	0	0	0	0	0	0	*151518*	*448232*
NNE	6	0	0	0	0	0	0	0	*154651*	*193313*
NE	0	0	0	0	0	0	0	0	*38663*	*96657*
ENE	0	0	0	0	0	0	828	0	0	0
E	0	0	0	0	0	0	2229	219	0	13561
ESE	0	0	320	0	0	0	11198	50152	20763	104445
SE	662	661	0	0	6786	27558	7443	9301	35612	11828
SSE	661	729	60	71	109	1593	2075	23880	6229	20419
S	4	12	55	328	651	1680	34083	7301	34694	7138
SSW	17	5	82	79	482	5743	4141	6025	26881	12565
SW	37	20	20	469	197	1728	9970	9130	7669	64607
WSW	0	50	0	35	84	1050	8246	12404	47735	14163
W	0	53	72	66	87	847	19318	259606	102087	25871
WNW	683	723	156	0	7274	4821	7009	207932	58896	13460
NW	0	165	595	0	0	1763	0	53092	20356	25771
NNW	20	138	0	0	0	0	0	20080	77289	233548

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.3-21	**E-98**	**Entire table**

In response to RAI 4.b, ER Table E.3-21, "Base Case Results for Internal Events at 50 Miles," is replaced in its entirety, and now reads:

Table E.3-21: Base Case Results for Internal Events at 50 Miles

Release Category	Whole Body Dose (50, rem)/yr	Economic Impact (50, $)/yr
1.1	4.91E-02	4.77E+01
1.2	3.07E-02	2.93E+01
1.3	1.37E+00	1.33E+03
1.4	3.66E-03	2.86E+00
2.1	3.25E-02	2.42E+01
2.2	5.56E-01	2.64E+02
3.1	2.20E-03	1.09E+00
3.2	1.35E-04	1.11E-01
3.3	2.16E-05	1.07E-02
3.4	1.23E-02	7.85E+00
4.1	3.73E-05	8.67E-03
4.2	3.57E-02	1.86E+01
4.3	7.01E-07	1.19E-04
4.4	1.08E-02	8.09E+00
5.1	9.77E-03	2.85E+00
5.2	1.32E-02	1.12E+01
5.3	9.41E-04	2.66E-01
5.4	7.36E-03	3.84E+00
6.1	5.50E-04	4.44E-01
6.2	6.07E-05	5.21E-02
6.3	4.01E-05	5.81E-03
6.4	1.90E-02	7.38E+00
7.1	5.63E-07	3.05E-05
7.2	7.35E-05	2.63E-02
7.3	5.37E-09	3.45E-07
7.4	8.09E-06	7.13E-04
7.5	3.75E-08	0.00E+00
7.6	6.57E-03	1.64E+00
7.7	2.90E-08	2.32E-07
7.8	1.92E-02	7.48E+00
8.1	1.20E-04	7.25E-04
8.2	1.01E-01	2.89E+01
9.1	2.03E-03	1.10E-04
9.2	2.09E-02	1.30E+00
Total	**2.30E+00**	**1.80E+03**

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.3-22	**E-99**	**Entire table**

In response to RAI 4.b, ER Table E.3-22, "Base Case Consequence Input to SAMA Analysis," is replaced in its entirety, and now reads:

Table E.3-22: Base Case Consequence Input to SAMA Analysis

Release Category	Whole Body Dose (50, rem)	Economic Impact (50, $)
1.1	2.23E+06	2.17E+09
1.2	2.36E+06	2.25E+09
1.3	2.32E+06	2.26E+09
1.4	3.05E+06	2.38E+09
2.1	5.41E+06	4.04E+09
2.2	1.03E+07	4.89E+09
3.1	8.81E+05	4.34E+08
3.2	4.83E+06	3.97E+09
3.3	8.63E+05	4.27E+08
3.4	7.22E+06	4.62E+09
4.1	3.73E+04	8.67E+06
4.2	1.05E+06	5.46E+08
4.3	6.37E+04	1.08E+07
4.4	1.40E+06	1.05E+09
5.1	3.37E+05	9.84E+07
5.2	3.47E+06	2.96E+09
5.3	3.36E+05	9.50E+07
5.4	8.27E+06	4.32E+09
6.1	1.25E+06	1.01E+09
6.2	1.84E+06	1.58E+09
6.3	8.91E+03	1.29E+06
6.4	6.12E+05	2.38E+08
7.1	4.02E+04	2.18E+06
7.2	1.29E+05	4.62E+07
7.3	2.44E+03	1.57E+05
7.4	3.37E+03	2.97E+05
7.5	1.39E+03	0.00E+00
7.6	3.46E+05	8.64E+07
7.7	8.05E+02	6.45E+03
7.8	1.96E+05	7.63E+07
8.1	1.90E+03	1.15E+04
8.2	7.79E+05	2.22E+08
9.1	2.67E+02	1.45E+01
9.2	1.49E+04	9.27E+05
Total	**5.97E+07**	**3.98E+10**

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Tables E.3-23 through E.3-32	**E-100 & E-101**	**Entire tables (10 tables)**

In response to RAI 4.b, ER Tables E.3-23 through E.3-32 are replaced in their entirety, and now read:

Table E.3-23: Comparison of Base Case and Case S1

	Internal Events		
	Base	**S1**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	2.30E+00	2.52E+00	9.6%
Economic Impact (50) ($/yr)	1.80E+03	1.96E+03	8.9%

Table E.3-24: Comparison of Base Case and Case S2

	Internal Events		
	Base	**S2**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	2.30E+00	2.05E+00	-10.9%
Economic Impact (50) ($/yr)	1.80E+03	1.61E+03	-10.6%

Table E.3-25: Comparison of Base Case and Case S3

	Internal Events		
	Base	**S3**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	2.30E+00	2.37E+00	3.0%
Economic Impact (50) ($/yr)	1.80E+03	1.80E+03	0.0%

Table E.3-26: Comparison of Base Case and Case M1

	Internal Events		
	Base	**M1**	**% diff.**
Whole Body Dose (50) (person-rem/yr)	2.30E+00	2.36E+00	2.6%
Economic Impact (50) ($/yr)	1.80E+03	1.81E+03	-0.6%

Table E.3-27: Comparison of Base Case and Case M2

	Internal Events		
	Base	M2	% diff.
Whole Body Dose (50) (person-rem/yr)	2.30E+00	2.20E+00	-4.3%
Economic Impact (50) ($/yr)	1.80E+03	1.78E+03	-1.1%

Table E.3-28: Comparison of Base Case and Case A1

	Internal Events		
	Base	A1	% diff.
Whole Body Dose (50) (person-rem/yr)	2.30E+00	1.75E+00	-23.9%
Economic Impact (50) ($/yr)	1.80E+03	1.42E+03	-21.1%

Table E.3-29: Comparison of Base Case and Case A2

	Internal Events		
	Base	A2	% diff.
Whole Body Dose (50) (person-rem/yr)	2.30E+00	0.30E+00	0.0%
Economic Impact (50) ($/yr)	1.80E+03	1.80E+03	0.0%

Table E.3-30: Comparison of Base Case and Case A3

	Internal Events		
	Base	A3	% diff.
Whole Body Dose (50) (person-rem/yr)	2.30E+00	2.31E+00	0.4%
Economic Impact (50) ($/yr)	1.80E+03	1.80E+03	0.0%

Table E.3-31: Comparison of Base Case and Case E1

	Internal Events		
	Base	E1	% diff.
Whole Body Dose (50) (person-rem/yr)	2.30E+00	2.28E+00	-0.9%
Economic Impact (50) ($/yr)	1.80E+03	1.80E+03	0.0%

Table E.3-32: Comparison of Base Case and Case E2

	Internal Events		
	Base	E2	% diff.
Whole Body Dose (50) (person-rem/yr)	2.30E+00	1.86E+00	-19.1%
Economic Impact (50) ($/yr)	1.80E+03	1.38E+03	-23.3%

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.3-33	**E-101**	**New table**

In response to RAI 6.j, new ER Table E.3-33, "Comparison of Base Case and Case E3," is added to the ER, which reads:

Table E.3-33: Comparison of Base Case and Case S1

	Internal Events		
	Base	***S1***	***% diff.***
Whole Body Dose (50) (person-rem/yr)	*2.30E+00*	*2.31E+00*	*0.4%*
Economic Impact (50) ($/yr)	*1.80E+03*	*1.80E+03*	*0.0%*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.4-1	**E-101**	**Entire table**

In response to RAIs 3.c and 4.b, ER Table E.4-1, "Total Cost of Severe Accident Impact," is replaced in its entirety, and now reads:

Table E.4-1: Total Cost of Severe Accident Impact

APE	$56,442
AOC	$22,086
AOE	$4,340
AOSC	$266,279
Severe Accident Impact (Internal Events)	$349,147
Fire, Seismic, Other	$1,606,076
Maximum Benefit (Internal Events, Fire, Seismic, Other)	$1,955,223

Affected LRA Section	LRA Page Nos.	Affected Paragraph and Sentence
ER Table E.5-3	**E-136 - 139**	**Entire table**

In response to RAI 2.e, ER Table E.5-3, "Basic Event LERF Importance," is replaced in its entirety, and now reads:

Table E.5-3: Basic Event LERF Importance

Event Name	F-V	RRW	Description
R	9.00E-01	10.031	Steam generator tube rupture <initiating event>
XHAMUCDE	6.04E-01	2.526	Operators fail to attempt cooldown via makeup/HPI cooling.
CHASGDPE	5.35E-01	2.151	Operators fail to cooldown during a steam generator tube rupture
LHAMSIVE	4.92E-01	1.970	Failure to close MSIV and isolate steam generator containing ruptured tube
AASGTR11	4.80E-01	1.925	SGTR occurs on OTSG 1-1 <split fraction>
AASGTR12	3.93E-01	1.647	SGTR occurs on OTSG 1-2 <split fraction>
FMM00003	7.88E-02	1.086	Any MSSVs on SG1 fail to reseat
VD-IEF	7.47E-02	1.081	ISLOCA due to internal rupture of DHR suction valves
FLCO101F	7.24E-02	1.078	Logic Card Fails during operation - MSIV 101 fails to close
LPPNISOZ	7.11E-02	1.077	ISLOCA occurs in non-isolable portion of DHR system
FMM00004	6.80E-02	1.073	Any MSSVs on SG2 fail to reseat
QHAMDFPE	6.80E-02	1.073	Failure to start MDFP as backup to turbine-driven feedwaer pumps for transient, Small LOCA or SGTR events
FLC0100F	6.07E-02	1.065	Logic Card Fails during operation - MSIV 100 fails to close
EC1ZXXXN-CC_1_2	5.18E-02	1.055	CCF of two components: EC1Z089N & EC1Z100N
LPSRC2BH	4.88E-02	1.051	Press switch PSH RC2B4 fails high – fails DHR
LPSZ416H	4.88E-02	1.051	Press switch PSH 7531A fails high - fails DHR
LMVF012R	4.49E-02	1.047	Internal rupture of DH 12 (annual frequency)
LMBCWRT1	4.09E-02	1.043	CWR Train 1 unavailable due to maintenance
EDG0012F	3.44E-02	1.036	EDG 1-2 fails to run

Table E.5-3: Basic Event LERF Importance (continued)			
Event Name	**F-V**	**RRW**	**Description**
FCIRCTMP	3.27E-02	1.034	Circ water temperature not acceptable
FVV011BT	3.02E-02	1.031	AVV ICS11B fails to reseat after steam release
LMVF011R	2.98E-02	1.031	Internal rupture of DH 11 (annual frequency)
ELOOPRT	2.91E-02	1.030	LOOP given reactor trip
EHAD2DGE	2.73E-02	1.028	Operators fail to align power from EDG 1-1 or EDG 1-2 to supply MDFP given LOOP
EHASBDGE	2.76E-02	1.028	Operators fail to align power from station blackout diesel generator to supply MDFP given LOOP
FVV011AT	2.60E-02	1.027	AVV ICS11A fails to reseat after steam release
LMVU011R	2.39E-02	1.024	Internal rupture of DH 11 since cold shutdown
LMVU012R	2.39E-02	1.024	Internal rupture of DH 12 since cold shutdown
LMBCWRT2	2.14E-02	1.022	CWR Train 2 unavailable due to maintenance
FLC011BF	1.95E-02	1.020	ICS logic card fails ICS11B (AVV SG1) fails to open
FLC011AF	1.83E-02	1.019	ICS logic card fails ICS11A (AVV SG2) fails to open
EC1Z100N	1.79E-02	1.018	Breaker HX11B fails to open – fails power from SU1 and SU2 to Bus B
EC1Z153C	1.79E-02	1.018	Breaker HX02B fails to close - fails power from SU1 to Bus B
XHOS-CCW1RUN2STBY	1.67E-02	1.017	CCW Pump 1 running, Pump 2 in standby
XHOS-CCW2RUN1STBY	1.65E-02	1.017	CCW Pump 2 running, Pump 1 in standby
EHASBD1E	1.61E-02	1.016	Operators fail to start SBODG and align to bus D1
ET4DF12F	1.53E-02	1.016	Transformer DF 1-2 local faults
LAV1761N	1.55E-02	1.016	Air-operated valve WC 1761 fails to open
EHAD1ACE	1.45E-02	1.015	Failure to lineup alternate source to bus D1
LMV0011H	1.50E-02	1.015	Motor-operated valve DH 11 fails to hold on high exposure
EB200D1F	1.30E-02	1.013	Bus D1 local faults not including fire
EDG0SBOF	1.31E-02	1.013	SBO diesel generator fails to run
LXV0125C	1.11E-02	1.011	Manual valve WC 125 fails to close – makeup to BWST for SGTR
LXV0169N	1.11E-02	1.011	Manual valve WC 169 fails to close – makeup to BWST for SGTR

Table E.5-3: Basic Event LERF Importance (continued)			
Event Name	**F-V**	**RRW**	**Description**
LXV0171C	1.11E-02	1.011	Manual valve WC 171 fails to close – makeup to BWST for SGTR
LXV0172C	1.11E-02	1.011	Manual valve WC 172 fails to close – makeup to BWST for SGTR
LXVBW15C	1.11E-02	1.011	Manual valve BW 15 fails to close – makeup to BWST for SGTR
LXVBW16N	1.11E-02	1.011	Manual valve BW 16 fails to close – makeup to BWST for SGTR
LXVSF79N	1.11E-02	1.011	Manual valve SF 79 fails to open – makeup to BWST for SGTR
LXVSF80C	1.11E-02	1.011	Manual valve SF 80 fails to open – makeup to BWST for SGTR
LXVSF87N	1.11E-02	1.011	Manual valve SF 87 fails to open – makeup to BWST for SGTR
LXVSF92C	1.11E-02	1.011	Manual valve SF 92 fails to close – makeup to BWST for SGTR
LXVWC44N	1.11E-02	1.011	Manual valve WC 44 fails to open – makeup to BWST for SGTR
EDG0SBOA	1.00E-02	1.010	SBO diesel generator fails to start
FIV0101C	1.02E-02	1.010	MS 101 (MSIV SG1) fails to close
VHAISOLR	1.02E-02	1.010	Operators fail to attempt to close DH1A to isolate ISLOCA
ZHAISOLR	1.02E-02	1.010	Failure to find and isolate ISLOCA resulting from reverse flow through LPI injection line
FIV0100C	8.43E-03	1.009	MS100 (MSIV SG2) fails to close
UHAMUHPE	8.89E-03	1.009	Failure to initiate makeup/HPI cooling after loss of all feedwater coincident with reactor trip
ZOP007BR	9.15E-03	1.009	Failure to recover offsite power within one hour to prevent loss of DC
EMBEDG12	7.76E-03	1.008	EDG Train 2 in maintenance
QMBAFP12	7.56E-03	1.008	AFW train 2 in maintenance
XHABWMUE	7.86E-03	1.008	Operators fail to initiate makeup to the BWST during a SGTR.
EB300F1F	6.47E-03	1.007	Bus F1 local faults
EDG0012A	6.55E-03	1.007	EDG 1-2 fails to start
EMBSBODG	7.22E-03	1.007	SBO diesel generator in maintenance
LMV0011N	7.02E-03	1.007	Motor-operated valve DH 11 fails to open
LMV0012N	7.02E-03	1.007	Motor-operated valve DH 12 fails to open
QMBAFP11	6.87E-03	1.007	AFW train 1 in maintenance
XHOS-AMB->40F	7.16E-03	1.007	Ambient temperature is > 40
EC1BET9N	6.03E-03	1.006	CCF for failure of 13.8 kV breakers to open
EC1CC09N	6.03E-03	1.006	Breaker HX11A OR HX11B fails to open
EC2Z012R	5.52E-03	1.006	Breaker AD1DF12 fails to remain closed

Table E.5-3: Basic Event LERF Importance (continued)			
Event Name	**F-V**	**RRW**	**Description**
LMV0011X	5.96E-03	1.006	Motor-operated valve DH 11 fails to close while indicating closed
LMV0012X	5.96E-03	1.006	Motor-operated valve DH 12 fails to close while indicating closed
VL10-IEF	6.39E-03	1.006	ISLOCA via Train 1 injection line reverse flow (initiating event)
VL20-IEF	6.41E-03	1.006	ISLOCA via Train 2 injection line reverse flow (initiating event)
EDG0011F	5.35E-03	1.005	EDG 1-1 fails to start
FMFWTRIP	4.70E-03	1.005	MFW/ICS faults following trip
LCVF030R	5.37E-03	1.005	Internal leak develops in check valve CF 30 (per year)
LCVF031R	5.35E-03	1.005	Internal leak develops in check valve CF 31 (per year)

Affected LRA Section	LRA Page Nos.	Affected Paragraph and Sentence
ER Table E.5-4	**E-144 - 154**	**6 rows revised; 1 new row**

In response to RAIs 5.c and 5.f, ER Table E.5-4, "List of Initial SAMA Candidates," is revised as follows:

Table E.5-4: List of Initial SAMA Candidates

SAMA Candidate Identifier	SAMA Candidate Description	Derived Benefit	Source
CB-21	Install pressure measurements between the two DHR suction valves in the line from the RCS hot leg.	This SAMA candidate would provide indication of failure of inboard isolation valves allowing time to initiate mitigating actions to prevent ISLOCA.	[2, Table 14] *[Table E.5-2]*
CC-19	Provide automatic switchover of HPI and LPI suction from the BWST to containment sump for LOCAs.	This SAMA candidate will increase the reliability of switchover of suction from the BWST to the containment sump by providing both manual and automatic switchover.	*[Table E.5-1]*
CP-19	Install a redundant containment fan system.	This SAMA candidate would increase containment heat removal ability. *SAMA candidate CP-19 was added as a variation to CP-18 to provide a redundant containment cooling function, in the form of containment fan coolers.*	*Davis-Besse containment cooling design*
CW-24	Replace the standby CCW pump with a pump diverse from the other two CCW pumps.	This SAMA candidate would improve CCW reliability by reducing the likelihood of a CCF of all three CCW pumps.	*[Table E.5-1]* *[Table E.5-2]*
CW-25	Provide the ability to cool make-up pumps using fire water in the event of loss of CCW.	This SAMA candidate would allow continued injection of RCP seal water in the event of loss of CCW.	*[Table E.5-1]* *[Table E.5-2]*
FW-16	Perform surveillances on manual valves used for backup AFW pump suction.	This SAMA candidate would improve the success probability for providing an alternate water supply to the AFW pumps.	~~*[2, Table 14]*~~ *[Table E.5-1]* *[Table E.5-2]*
OT-09R	*Provide operator training with PRA-identified high risk important human actions to be emphasized in training.*	*PRA results show that operator actions are significant contributors to overall plant risk. By highlighting those operator actions shown to have the highest risk importance, the reliability of those actions will be improved.*	*Table E.5-2*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.6-1	**E-155 – E-180**	**6 rows revised; 1 new row**

In response to RAIs 5.c, 5.g, 5.h, 6.b, and 6.k, ER Table E.6-1, "Qualitative Screening of SAMA Candidates," is revised as follows:

Table E.6-1: Qualitative Screening of SAMA Candidates

SAMA ID	Modification (Potential Enhancement)	Screening Criteria	Basis for Screening/Modification Enhancements
CB-08	Improve operator training on ISLOCA coping.	~~*Criterion E*~~ ~~*Subsumed*~~ *Criterion B* *Already Implemented*	~~*This SAMA would reduce the risk of ISLOCA events by improving the likelihood of timely identification and diagnosis of ISLOCA events and thereby increasing the likelihood of successful mitigating actions. This SAMA will be subsumed in CB-07.*~~ *Davis-Besse has several procedures in place to address small and interfacing system LOCAs. Operators receive training on LOCAs, and there are a number of indications to support the likelihood and timely identification and diagnosis of ISLOCA events (including tank level indications, lifting relief valves, and running sump pumps).*
CC-08	Add the ability to automatically align ECCS to recirculation mode upon BWST depletion.	Criterion E Subsumed	Davis-Besse currently has the ability to initiate automatic switchover from the BWST to the containment sump on low BWST level, but this feature has been deactivated. ~~*The cost would by minor to reactivate this feature.*~~ This SAMA candidate will be subsumed in SAMA candidate CC-19.
CC-19	Provide automatic switchover of HPI and LPI suction from the BWST to containment sump for LOCAs.	Criterion F Considered for Further Evaluation	Davis-Besse currently has the ability to initiate automatic switchover from the BWST to the containment sump on low BWST level, but this feature has been deactivated. ~~*The cost would by minor to reactivate this feature.*~~ Therefore, this SAMA candidate is considered for further evaluation.

Table E.6-1: Qualitative Screening of SAMA Candidates (continued)			
SAMA ID	**Modification (Potential Enhancement)**	**Screening Criteria**	**Basis for Screening/Modification Enhancements**
FL-01	Improve inspection of rubber expansion joints on main condenser.	~~*Criterion D*~~ ~~*Very Low Benefit*~~ *Criterion B* *Already Implemented*	~~*Based on the top 100 cutsets and component basic event importance, circulating water breaks are not a significant risk contributor at Davis-Besse.*~~ *The circulating water joints are currently inspected during outages, and include both interior and exterior inspections. Exterior inspections of the visible portion of the expansion joint are performed during Engineering system walkdowns and Operator tours. Additionally, the expansion joints are periodically replaced.*
OT-05	Increase training and operating experience feedback to improve operator response.	~~*Criterion D*~~ ~~*Very Low Benefit*~~ *Criterion B* *Already Implemented*	~~*No deficiencies in operator training or feedback are identified.*~~ *FENOC provides PRA information, such as risk-significant initiating events, high worth operator actions and high worth equipment, to various departments, including Operations Training, and presents this information on posters throughout the plant.*
OT-07	Install secondary side guard pipes up to the MSIVs.	Criterion D Very Low Benefit	*Steam line breaks are not a significant contributor to CDF or LERF.* The derived benefit would not justify the implementation cost required.
OT-09R	*Provide operator training with PRA-identified high risk important human actions to be emphasized in training.*	*Criterion B* *Already Implemented*	*Davis-Besse provides PRA information such as risk significant initiating events, high worth operator actions and high worth equipment. This information is provided to various departments and is presented on posters throughout the plant.*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
ER Table E.7-2	**E-183 - 185**	**Entire table**
ER Table E.7-3	**E-186**	**Entire table**
ER Table E.7-5	**E-188**	**Entire table**
ER Table E.8-1	**E-189 - 190**	**Entire table**

In response to RAIs 3.c and 4.b, ER Tables E.7-2, E.7-3, E.7-5 and E.8-1 are replaced in their entirety, and now read as shown on the following pages:

Table E.7-2: Internal Events Benefit Results for Analysis Case

Case	Maximum Benefit	AC/DC-01 (DCBattery)	AC/DC-03 (Battery Charger)	AC/DC-14 (GasTurbineGen)
Off-site Annual Dose (rem)	2.30E+00	2.28E+00	2.07E+00	2.05E+00
Off-site Annual Property Loss ($)	$1,800	$1,790	$1,610	$1,650
Comparison CDF[4]	----	1.0E-05	1.0E-05	1.0E-05
Comparison Dose (rem)	----	2.30E+00	2.30E+00	2.30E+00
Comparison Cost ($)	----	$1,800	$1,800	$1,800
Enhanced CDF	----	9.4E-06	7.8E-06	9.0E-06
Reduction in CDF	----	**6.00%**	**22.00%**	**10.00%**
Reduction in Off-site Dose	----	**0.87%**	**10.00%**	**10.87%**
Immediate Dose Savings (On-site)	$810	$49	$178	$81
Long Term Dose Savings (On-site)	$3,530	$212	$777	$353
Total Accident Related Occupational Exposure **(AOE)**	$4,340	$260	$955	$434
Cleanup/Decontamination Savings (On-site)	$132,362	$7,942	$29,120	$13,236
Replacement Power Savings (On-site)	$133,917	$8,035	$29,462	$13,392
Averted Costs of On-site Property Damage **(AOSC)**	$266,279	$15,977	$58,581	$26,628
Total On-site Benefit	**$270,619**	**$16,237**	**$59,536**	**$27,062**
Averted Public Exposure **(APE)**	$56,442	$491	$5,644	$6,135
Averted Off-site Damage Savings **(AOC)**	$22,086	$123	$2,331	$1,841
Total Off-site Benefit	**$78,528**	**$614**	**$7,976**	**$7,976**
Total Benefit (On-site + Off-site)	**$349,147**	**$16,851**	**$67,512**	**$35,037**

[4] The sum of the Containment Systems State frequencies calculated by the Level 2 PRA model is slightly different than the CDF calculated by the Level 1 PRA due to the delete term approximation and the additional systems included in the Level 2 models.

Table E.7-2: Internal Events Benefit Results for Analysis Case (continued)

Case	AC/DC-19 (FireWaterBackup)	AC/DC-21 (RepairBreakers)	AC/DC-25 (DedDCPower)	AC/DC-26 (Generator_TDAFW)
Off-site Annual Dose (rem)	2.28E+00	2.29E+00	2.25E+00	2.25E+00
Off-site Annual Property Loss ($)	$1,790	$1,790	$1,780	$1,780
Comparison CDF[4]	1.0E-05	1.0E-05	1.0E-05	1.0E-05
Comparison Dose (rem)	2.30E+00	2.30E+00	2.30E+00	2.30E+00
Comparison Cost ($)	$1,800	$1,800	$1,800	$1,800
Enhanced CDF	9.8E-06	9.7E-06	8.5E-06	8.5E-06
Reduction in CDF	**2.00%**	**3.00%**	**15.00%**	**15.00%**
Reduction in Off-site Dose	**0.87%**	**0.43%**	**2.17%**	**2.17%**
Immediate Dose Savings (On-site)	$16	$24	$121	$121
Long Term Dose Savings (On-site)	$71	$106	$529	$529
Total Accident Related Occupational Exposure **(AOE)**	$87	$130	$651	$651
Cleanup/Decontamination Savings (On-site)	$2,647	$3,971	$19,854	$19,854
Replacement Power Savings (On-site)	$2,678	$4,018	$20,088	$20,088
Averted Costs of On-site Property Damage **(AOSC)**	$5,326	$7,988	$39,942	$39,942
Total On-site Benefit	**$5,412**	**$8,119**	**$40,593**	**$40,593**
Averted Public Exposure **(APE)**	$491	$245	$1,227	$1,227
Averted Off-site Damage Savings **(AOC)**	$123	$123	$245	$245
Total Off-site Benefit	**$614**	**$368**	**$1,472**	**$1,472**
Total Benefit (On-site + Off-site)	**$6,026**	**$8,487**	**$42,065**	**$42,065**

[4] The sum of the Containment Systems State frequencies calculated by the Level 2 PRA model is slightly different than the CDF calculated by the Level 1 PRA due to the delete term approximation and the additional systems included in the Level 2 models.

Table E.7-2: Internal Events Benefit Results for Analysis Case (continued)

Case	AC/DC-27 (SBO_DieselTank)	CB-21 (DHR_valves)	CC-01 (HPI_System)	CC-04 (LPI_pump)
Off-site Annual Dose (rem)	2.30E+00	2.11E+00	2.29E+00	2.30E+00
Off-site Annual Property Loss ($)	$1,800	$1,710	$1,790	$1,800
Comparison CDF[4]	1.0E-05	1.0E-05	1.0E-05	1.0E-05
Comparison Dose (rem)	2.30E+00	2.30E+00	2.30E+00	2.30E+00
Comparison Cost ($)	$1,800	$1,800	$1,800	$1,800
Enhanced CDF	1.0E-05	1.0E-05	1.0E-05	1.0E-05
Reduction in CDF	**0.00%**	**0.00%**	**0.00%**	**0.00%**
Reduction in Off-site Dose	**0.00%**	**8.26%**	**0.43%**	**0.00%**
Immediate Dose Savings (On-site)	$0	$0	$0	$0
Long Term Dose Savings (On-site)	$0	$0	$0	$0
Total Accident Related Occupational Exposure **(AOE)**	$0	$0	$0	$0
Cleanup/Decontamination Savings (On-site)	$0	$0	$0	$0
Replacement Power Savings (On-site)	$0	$0	$0	$0
Averted Costs of On-site Property Damage **(AOSC)**	$0	$0	$0	$0
Total On-site Benefit	**$0**	**$0**	**$0**	**$0**
Averted Public Exposure **(APE)**	$0	$4,663	$245	$0
Averted Off-site Damage Savings **(AOC)**	$0	$1,104	$123	$0
Total Off-site Benefit	**$0**	**$5,767**	**$368**	**$0**
Total Benefit (On-site + Off-site)	**$0**	**$5,767**	**$368**	**$0**

[4] The sum of the Containment Systems State frequencies calculated by the Level 2 PRA model is slightly different than the CDF calculated by the Level 1 PRA due to the delete term approximation and the additional systems included in the Level 2 models.

Table E.7-2: Internal Events Benefit Results for Analysis Case (continued)

Case	CC-05 (LPI_Diesel_pump)	CC-19 (BWST_to_Sump)	HV-01 (Redundant_HVAC)	HV-03 (Backup_fans)
Off-site Annual Dose (rem)	2.30E+00	2.30E+00	2.30E+00	2.30E+00
Off-site Annual Property Loss ($)	$1,800	$1,800	$1,800	$1,800
Comparison CDF[4]	1.0E-05	1.0E-05	1.0E-05	1.0E-05
Comparison Dose (rem)	2.30E+00	2.30E+00	2.30E+00	2.30E+00
Comparison Cost ($)	$1,800	$1,800	$1,800	$1,800
Enhanced CDF	1.0E-05	9.9E-06	1.0E-05	1.0E-05
Reduction in CDF	**0.00%**	**1.00%**	**0.00%**	**0.00%**
Reduction in Off-site Dose	**0.00%**	**0.00%**	**0.00%**	**0.00%**
Immediate Dose Savings (On-site)	$0	$8	$0	$0
Long Term Dose Savings (On-site)	$0	$35	$0	$0
Total Accident Related Occupational Exposure **(AOE)**	$0	$43	$0	$0
Cleanup/Decontamination Savings (On-site)	$0	$1,324	$0	$0
Replacement Power Savings (On-site)	$0	$1,339	$0	$0
Averted Costs of On-site Property Damage **(AOSC)**	$0	$2,663	$0	$0
Total On-site Benefit	**$0**	**$2,706**	**$0**	**$0**
Averted Public Exposure **(APE)**	$0	$0	$0	$0
Averted Off-site Damage Savings **(AOC)**	$0	$0	$0	$0
Total Off-site Benefit	**$0**	**$0**	**$0**	**$0**
Total Benefit (On-site + Off-site)	**$0**	**$2,706**	**$0**	**$0**

[4] The sum of the Containment Systems State frequencies calculated by the Level 2 PRA model is slightly different than the CDF calculated by the Level 1 PRA due to the delete term approximation and the additional systems included in the Level 2 models.

Table E.7-3: Total Benefit Results for Analysis Cases

	Maximum Benefit	AC/DC-01 (DCBattery)	AC/DC-03 (Battery Charger)	AC/DC-14 (GasTurbineGen)	AC/DC-19 (FireWaterBackup)	AC/DC-21 (RepairBreakers)	AC/DC-25 (DedDCPower)
Internal Events	$349,147	$16,851	$67,512	$35,037	$6,026	$8,487	$42,065
Fires, Seismic, Other	$1,606,076	$77,513	$310,553	$161,172	$27,719	$39,039	$193,500
Total Benefit	**$1,955,223**	**$94,363**	**$378,065**	**$196,209**	**$33,745**	**$47,525**	**$235,565**

	AC/DC-26 (Generator_TDAFW)	AC/DC-27 (SBO_DieselTank)	CB-21 (DHR_valves)	CC-01 (HPI_System)	CC-04 (LPI_pump)	CC-05 (LPI_Dieselpump)	CC-19 (BWST_to_Sump)
Internal Events	$42,065	$0	$5,767	$368	$0	$0	$2,706
Fires, Seismic, Other	$193,500	$0	$26,528	$1,693	$0	$0	$12,448
Total Benefit	**$235,565**	**$0**	**$32,295**	**$2,061**	**$0**	**$0**	**$15,155**

	HV-01 (Redundant_HVAC)	HV-03 (Backup_fans)
Internal Events	$0	$0
Fires, Seismic, Other	$0	$0
Total Benefit	**$0**	**$0**

Table E.7-5: Final Results of Cost Benefit Evaluation

SAMA Candidate ID	Modification	Estimated Benefit	2009 Estimate Cost	Conclusion
AC/DC-01	Provide additional DC battery capacity.	$94,363	$1,750,000	Not Cost Effective
AC/DC-03	Add a portable, diesel-driven battery charger to existing DC system.	$378,065	$330,000	Cost Effective
AC/DC-14	Install a gas turbine generator.	$196,209	$2,000,000	Not Cost Effective
AC/DC-19	Use fire water system as a backup source for diesel cooling.	$33,745	$700,000	Not Cost Effective
AC/DC-21	Develop procedures to repair or replace failed 4kV breakers.	$47,525	$100,000	Not Cost Effective
AC/DC-25	Provide a dedicated DC power system (battery/battery charger) for the TDAFW control valve and NNI-X for steam generator level indication.	$235,565	$2,000,000	Not Cost Effective
AC/DC-26	Provide an alternator/generator that would be driven by each TDAFW pump.	$235,565	$2,000,000	Not Cost Effective
AC/DC-27	Increase the size of the SBO fuel oil tank.	$0	$550,000	Not Cost Effective
CB-21	Install pressure measurements between the two DHR suction valves in the line from the RCS hot leg.	$32,295	$550,000	Not Cost Effective
CC-01	Install an independent active or passive HPI system.	$2,061	$6,500,000	Not Cost Effective
CC-04	Add a diverse LPI system.	$0	$5,500,000	Not Cost Effective
CC-05	Provide capability for alternate LPI via diesel-driven fire pump.	$0	$6,500,000	Not Cost Effective
CC-19	Provide automatic switchover of HPI and LPI suction from the BWST to containment sump for LOCAs.	$15,155	$1,500,000	Not Cost Effective
HV-01	Provide a redundant train or means of ventilation.	$0	$50,000	Not Cost Effective
HV-03	Stage backup fans in switchgear rooms.	$0	$400,000	Not Cost Effective

Table E.8-1: Final Results of the Sensitivity Cases

SAMA Candidate ID	Repair Case	Low Discount Rate Case	High Discount Rate Case	On-site Dose Case	On-site Cleanup Case	Replacement Power Case	2009 Estimated Cost	Conclusion
AC/DC-01	$58,367	$142,486	$64,929	$95,839	$109,188	$124,566	$1,750,000	Not Cost Effective
AC/DC-03	$246,076	$571,954	$262,617	$383,474	$432,421	$488,806	$330,000	Cost Effective
AC/DC-14	$136,214	$297,589	$138,018	$198,668	$220,917	$246,546	$2,000,000	Not Cost Effective
AC/DC-19	$21,746	$51,031	$23,396	$34,237	$38,686	$43,812	$700,000	Not Cost Effective
AC/DC-21	$29,527	$71,774	$32,727	$48,263	$54,938	$62,626	$100,000	Not Cost Effective
AC/DC-25	$145,573	$355,685	$162,059	$239,253	$272,626	$311,071	$2,000,000	Not Cost Effective
AC/DC-26	$145,573	$355,685	$162,059	$239,253	$272,626	$311,071	$2,000,000	Not Cost Effective
AC/DC-27	$0	$0	$0	$0	$0	$0	$550,000	Not Cost Effective
CB-21	$32,295	$49,858	$24,719	$32,295	$32,295	$32,295	$550,000	Not Cost Effective
CC-01	$2,061	$3,182	$1,578	$2,061	$2,061	$2,061	$6,500,000	Not Cost Effective
CC-04	$0	$0	$0	$0	$0	$0	$5,500,000	Not Cost Effective
CC-05	$0	$0	$0	$0	$0	$0	$6,500,000	Not Cost Effective
CC-19	$9,155	$22,864	$10,383	$15,401	$17,625	$20,188	$1,500,000	Not Cost Effective
HV-01	$0	$0	$0	$0	$0	$0	$50,000	Not Cost Effective
HV-03	$0	$0	$0	$0	$0	$0	$400,000	Not Cost Effective

Table E.8-1: Final Results of the Sensitivity Cases (continued)

SAMA Candidate ID	Multiplier Case	Evacuation Speed	Off-site Economic Cost	2009 Estimated Cost	Conclusion
AC/DC-01	$134,805	$125,284	$95,738	$1,750,000	Not Cost Effective
AC/DC-03	$540,092	$408,985	$379,439	$330,000	Cost Effective
AC/DC-14	$280,299	$227,130	$197,583	$2,000,000	Not Cost Effective
AC/DC-19	$48,207	$64,665	$35,119	$700,000	Not Cost Effective
AC/DC-21	$67,893	$78,446	$48,899	$100,000	Not Cost Effective
AC/DC-25	$336,521	$266,485	$236,939	$2,000,000	Not Cost Effective
AC/DC-26	$336,521	$266,485	$236,939	$2,000,000	Not Cost Effective
AC/DC-27	$0	$30,920	$1,374	$550,000	Not Cost Effective
CB-21	$46,135	$63,215	$33,669	$550,000	Not Cost Effective
CC-01	$2,945	$32,982	$3,436	$6,500,000	Not Cost Effective
CC-04	$0	$30,920	$1,374	$5,500,000	Not Cost Effective
CC-05	$0	$30,920	$1,374	$6,500,000	Not Cost Effective
CC-19	$21,649	$46,075	$16,529	$1,500,000	Not Cost Effective
HV-01	$0	$30,920	$1,374	$50,000	Not Cost Effective
HV-03	$0	$30,920	$1,374	$400,000	Not Cost Effective

ATTACHMENT 7

Letter from B. Allen, FirstEnergy, to NRC Document Control Desk, "Reply to Supplemental Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4613) Environmental Report Severe Accident Mitigation Alternatives Analysis" (Sept. 1, 2011)

FENOC
FirstEnergy Nuclear Operating Company

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Vice President - Nuclear

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September 1, 2011
L-11-251

10 CFR 54

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Davis-Besse Nuclear Power Station, Unit No. 1
Docket No. 50-346, License Number NPF-3
Reply to Supplemental Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application, (TAC No. ME4613) Environmental Report Severe Accident Mitigation Alternatives Analysis

By letter dated August 27, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102450565), FirstEnergy Nuclear Operating Company (FENOC) submitted an application pursuant to Title 10 of the *Code of Federal Regulations*, Part 54 for renewal of Operating License NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS). During a telephone conference call on July 29, 2011, with Ms. Paula Cooper, Nuclear Regulatory Commission (NRC) Environmental Project Manager, the NRC discussed supplemental requests for additional information (RAIs) to clarify FENOC responses to the severe accident mitigation alternatives (SAMA) analysis RAIs submitted by FENOC letter dated June 24, 2011 (ADAMS Accession No. ML11180A233). FENOC agreed to submit responses to the NRC supplemental SAMA analysis RAIs discussed during the call.

The Attachment provides the FENOC response to the NRC supplemental RAIs. The NRC request is shown in bold text followed by the FENOC response.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Clifford I. Custer, Fleet License Renewal Project Manager, at 724-682-7139.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 1, 2011.

Sincerely,

Barry S. Allen

Attachment:
Reply to Supplemental Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS), License Renewal Application, Environmental Report, Attachment E, Severe Accident Mitigation Alternatives (SAMA) Analysis

cc: NRC DLR Project Manager
NRC DLR Environmental Project Manager
NRC Region III Administrator

cc: w/o Attachment or Enclosure
NRC DLR Director
NRR DORL Project Manager
NRC Resident Inspector
Utility Radiological Safety Board

Supplemental Question RAI 1.[d]

Clarify whether the scope of the 2008 "gap self assessment" included Level 2 as well as Level 1 internal events, and whether a review of internal flooding and the high winds hazard was performed.

SUPPLEMENTAL RESPONSE RAI 1.d

The 2008 gap self assessment

- included Level 2 as well as Level 1 internal events;
- did not include internal flooding; and,
- did not include high winds.

Supplemental Question RAI 4.b

Was the escalation factor for the updated analysis the same one as used originally, and if not what was it? Was transient population considered in the updated analysis?

SUPPLEMENTAL RESPONSE RAI 4.b

The population escalation factor used for the updated analysis (accounting for the Canadian population) is the same as was used for the original analysis.

Transient population (between 0-30 miles) was considered in the original analysis. The same transient population was considered in the updated analysis.

Supplemental Question RAI 5.b

Clarify which and how applicable SAMAs meet the intent of improving seismic capacity for the BWST. Address in your response the cited SAMAs (i.e., AC/DC-01, CC-10, and CW-09) and SAMA CC-19.

SUPPLEMENTAL RESPONSE RAI 5.b

SAMA candidate CC-10 considers providing an in-containment reactor water storage tank; this SAMA candidate would meet the intent of improving seismic capacity for the borated water storage tank (BWST) by providing a suction source to the injection pumps independent of the BWST.

SAMA candidates AC/DC-01 (provide additional DC battery capacity) and CW-09 (additional training on loss of component cooling water) are not related to the BWST. SAMA candidate CC-19 addresses switchover from the BWST to the containment sump, which does not meet the intent of improving seismic capacity.

Supplemental Question RAI 5.d

1. **Clarify whether the automatic actions that were identified and evaluated in the response to RAI 5.d.ii are the only candidates or meant to be representative of other possibilities. Clarify that further unevaluated potentially cost beneficial automating options do not remain.**

2. **Describe the PRA modeling assumptions used to calculate the SAMA benefits for AC/DC-[28R] and OT-08R similar to that shown in Table E.7-1 of the ER.**

SUPPLEMENTAL RESPONSE RAI 5.d

1. The following SAMA candidates evaluate automating operator actions. Only those SAMA candidates that were evaluated in detail are listed here; SAMA candidates that were screened (or subsumed, or already implemented) are not listed even if they considered automating operator actions.

    - AC/DC-14 (Table E.7-1) – makes the station blackout diesel generator and corresponding human failure event perfectly reliable.

    - AC/DC-25 (Table E.7-1) – provides dedicated DC power for auxiliary feedwater pump control and eliminates the need for local manual control.

- AC/DC-26 (Table E.7-1) – provides an alternator/generator driven by the auxiliary feedwater pumps to provide DC power for the auxiliary feedwater pumps and eliminates the need for local manual control.

- AC/DC-27 (Table E.7-1) – makes the human failure event to refuel the station blackout diesel generator fuel tank perfectly reliable.

- AC/DC-28R (RAI 5.d) – automatically starts and loads the station blackout diesel generator on Bus D2 upon loss of power to the bus.

- CC-19 (Table E.7-1) – makes the human failure events for switchover of high pressure injection and low pressure injection suction from the BWST to the containment sump for loss of coolant accidents perfectly reliable.

- CC-22R (RAI 7.d) – automates refill of the BWST.

- CW-26R (RAI 7.a) – automates reactor coolant pump trip on high motor bearing cooling temperature.

- FW-17R (RAI 7.e) – automates start of the motor-driven feedwater pump in the event the automated emergency feedwater system is unavailable.

- OT-08R (RAI 5.d) - automatically starts and loads the station blackout diesel generator on Bus D2 upon loss of power to the bus in combination with automatically starting the motor-driven feedwater pump.

As described in the FENOC response (ML11180A233) to RAI 5.c, internal events and large early release frequency (LERF) basic events (including human failure events) with a risk reduction worth (RRW) equal to or greater than the cost of a procedure change were identified and evaluated. Hardware modifications were also considered based on RRW values. This method was judged to identify all potentially cost-beneficial automating options.

2. The probabilistic risk assessment (PRA) modeling assumptions used to calculate the SAMA benefits for SAMA candidates AC/DC-28R and OT-08R are as follows:

   - SAMA candidate AC/DC-28R evaluates automatically starting and loading the Davis-Besse station blackout diesel generator on Bus D2 upon loss of power to the bus.

     A bounding assessment of the potential benefit of automatically starting the station blackout diesel generator and loading it on bus D2 upon loss of power to the bus was performed by removing the human action to start the station blackout diesel generator from the cutsets.

     Core damage frequency (CDF) = 8.17E-06/yr.

- SAMA candidate OT-08R evaluates automatically starting and loading the station blackout diesel generator on Bus D2 upon loss of power to the bus in combination with automatically starting the motor-driven feedwater pump.

  A bounding assessment of the potential benefit of automatically starting the station blackout diesel generator and loading it on bus D2 upon loss of power to the bus with automatically starting the motor-driven feedwater pump was performed by removing the human actions to start the station blackout diesel generator and motor-driven feedwater pump from the cutsets.

  CDF = 5.43E-06/yr.

Supplemental Question RAI 6.j

Provide the increased evacuation speed used in the Case E1 sensitivity analysis.

SUPPLEMENTAL RESPONSE RAI 6.j

The increased evacuation speed used in sensitivity case E1 (use a more realistic (higher) speed of evaluation (ESPEED)) is 1.0 meters/second.

Supplemental Question RAI 7.a - 7.f

Describe the PRA modeling assumptions used to calculate the SAMA benefit similar to that shown in Table E.7-1 of the ER.

SUPPLEMENTAL RESPONSE RAI 7.a - 7.f

7.a

A SAMA candidate (CW-26R) to provide an automatic reactor coolant pump trip on loss of cooling to the reactor coolant pump seal thermal barrier cooler and loss of seal injection flow was evaluated for Davis-Besse.

A bounding assessment of the potential benefit of automating a reactor coolant pump trip on high motor bearing cooling temperature or on a loss of cooling to the reactor coolant pump seal thermal barrier cooler and a loss of seal injection flow was performed

by making the operator action to trip the reactor coolant pumps on loss of seal cooling and injection perfectly reliable.

CDF = 7.50E-06/yr.

7.b

As described in the FENOC response (ML11180A233) to RAI 7.b, the Davis-Besse design and PRA already includes use of the Decay Heat Removal System as a suction source for high pressure injection. For cases in which reactor coolant system pressure is too high for adequate flow, the high pressure injection pumps can be aligned to take suction from the discharge of the decay heat removal pumps; this is possible with the BWST as the suction source or with the containment sump as the suction source.

7.c

As described in the FENOC response (ML11180A233) to RAI 7.c, this SAMA candidate considers automating high pressure injection on low pressurizer level following a loss of secondary side heat removal where Reactor Coolant System pressure remains high while level drops. This SAMA was a viable consideration for Three Mile Island (TMI) based on plant design and system configuration. At TMI, the High Pressure Injection System is also the makeup system – there is a single Makeup and Purification System that provides normal makeup as well as standby Engineered Safety Actuation Signal (ESAS)-selected pumps which automatically inject high-pressure water into the Reactor Coolant System from the BWST in mitigation of loss of coolant accident scenarios. In addition, as discussed in Volume 3 of the Babcock and Wilcox Emergency Operating Procedure Technical Basis Document (EOP TBD), (Chapter III.C, “Lack of Adequate Primary to Secondary Heat Transfer”), for all plants except Davis-Besse, high pressure injection cooling must not be intentionally delayed if feedwater is not available. High pressure injection cooling must be established in a timely manner to assure adequate core cooling; it must be started early enough to slow Reactor Coolant System inventory depletion so that high pressure injection cooling will match decay heat before the core is uncovered.

At Davis-Besse, however, the plant design and systems are different from those at TMI. Davis-Besse has a separate safety High Pressure Injection System in addition to the normally-operating makeup system. The Davis-Besse High Pressure Injection System is not capable of injecting water into the RCS until pressure reaches ~1600 psig. In addition, because Davis-Besse has two makeup pumps, makeup/high pressure injection cooling can be delayed until the core outlet temperature reaches 600°F provided the Reactor Coolant System pressure-temperature limit is not exceeded. Although the Davis-Besse PRA considers makeup/ high pressure injection cooling in response to a loss of feedwater, including the associated operator actions, automating this function was not considered because of the complexity associated with the number of options and systems involved (e.g., pumps, valves, and alignment options, injection line options, and bleed options). Consequently, this SAMA candidate was not considered for Davis-Besse.

7.d

A SAMA candidate (CC-22R) to provide an automatic refill of the BWST was evaluated for Davis-Besse.

A bounding assessment of the potential benefit of automating refill of the BWST was performed by making the operator action to refill the BWST perfectly reliable.

CDF = 9.76E-06/yr.

7.e

A SAMA candidate (FW-17R) to automatically start the auxiliary feedwater pump when the emergency feedwater system is unavailable was evaluated for Davis-Besse. Based on the Davis-Besse design, this SAMA was interpreted as automatically starting the motor-driven feedwater pump in the event both turbine-driven auxiliary feedwater pumps were not available.

A bounding assessment of the potential benefit of automating start of the motor-driven feedwater pump was performed by removing cutsets containing operator actions to start the motor-driven feedwater pump, thereby making the operator actions to start the motor-driven feedwater pump perfectly reliable.

CDF = 7.03E-06/yr.

7.f

A SAMA candidate (CB-22R) to use a “gagging” device that could be used to close a stuck-open steam generator safety valve for a steam generator tube rupture was evaluated for Davis-Besse.

A bounding assessment of the potential benefit of utilizing a “gagging device” on a stuck open main steam safety valve was performed by removing main steam safety valve failures to close from the cutsets.

CDF = 9.24E-06/yr.

ATTACHMENT 8

NUREG-1465, *Accident Source Terms for Light-Water Nuclear Power Plants* (Feb. 1995)

NUREG–1465

Accident Source Terms for Light-Water Nuclear Power Plants

Final Report

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

L. Soffer, S. B. Burson, C. M. Ferrell,
R. Y. Lee, J. N. Ridgely

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Accident Source Terms for Light-Water Nuclear Power Plants

Final Report

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Abstract

In 1962 The U.S. Atomic Energy Commission published TID-14844, "Calculation of Distance Factors for Power and Test Reactors" which specified a release of fission products from the core to the reactor containment in the event of a postulated accident involving a "substantial meltdown of the core." This "source term," the basis for the NRC's Regulatory guides 1.3 and 1.4, has been used to determine compliance with the NRC's reactor site criteria, 10 CFR Part 100, and to evaluate other important plant performance requirements. During the past 30 years substantial additional information on fission product releases has been developed based on significant severe accident research. This document utilizes this research by providing more realistic estimates of the "source term" release into containment, in terms of timing, nuclide types, quantities, and chemical form, given a severe core-melt accident. This revised "source term" is to be applied to the design of future Light Water Reactors (LWRs). Current LWR licensees may voluntarily propose applications based upon it. These will be reviewed by the NRC staff.

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Preface

In 1962, the Atomic Energy Commission issued Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactors." In this document, a release of fission products from the core of a light-water reactor (LWR) into the containment atmosphere ("source term") was postulated for the purpose of calculating off-site doses in accordance with 10 CFR Part 100, "Reactor Site Criteria." The source term postulated an accident that resulted in substantial meltdown of the core, and the fission products assumed released into the containment were based on an understanding at that time of fission product behavior. In addition to site suitability, the regulatory applications of this source term (in conjunction with the dose calculation methodology) affect the design of a wide range of plant systems.

In the past 30 years, substantial information has been developed updating our knowledge about severe LWR accidents and the resulting behavior of the released fission products. The purpose of this document is to provide a postulated fission product source term released into containment that is based on current understanding of LWR accidents and fission product behavior. The information contained in this document is applicable to LWR designs and is intended to form the basis for the development of regulatory guidance, primarily for future LWRs. This report will serve as a basis for possible changes to regulatory requirements. However, acceptance of any proposed changes will be on a case-by-case basis.

Source terms for future reactors may differ from those presented in this report which are based upon insights derived from current generation light-water reactors. An applicant may propose changes in source term parameters (timing, release magnitude, and chemical form) from those contained in this report, based upon and justified by design specific features.

1 INTRODUCTION AND BACKGROUND

1.1 Regulatory Use of Source Terms

The use of postulated accidental releases of radioactive materials is deeply embedded in the regulatory policy and practices of the U.S. Nuclear Regulatory Commission (NRC). For over 30 years, the NRC's reactor site criteria in 10 CFR Part 100 (Ref. 1) have required, for licensing purposes, that an accidental fission product release resulting from "substantial meltdown" of the core into the containment be postulated to occur and that its potential radiological consequences be evaluated assuming that the containment remains intact but leaks at its maximum allowable leak rate. Radioactive material escaping from the containment is often referred to as the "radiological release to the environment." The radiological release is obtained from the containment leak rate and a knowledge of the airborne radioactive inventory in the containment atmosphere. The radioactive inventory within containment is referred to as the "in-containment accident source term."

The expression "in-containment accident source term," as used in this document, denotes the radioactive material composition and magnitude, as well as the chemical and physical properties of the material within the containment that are available for leakage from the reactor to the environment. The "in-containment accident source term" will normally be a function of time and will involve consideration of fission products being released from the core into the containment as well as removal of fission products by plant features intended to do so (e.g., spray systems) or by natural removal processes.

For currently licensed plants, the characteristics of the fission product release from the core into the containment are set forth in Regulatory Guides 1.3 and 1.4 (Refs. 2,3) and have been derived from the 1962 report, TID-14844 (Ref. 4). This release consists of 100% of the core inventory of noble gases and 50% of the iodines (half of which are assumed to deposit on interior surfaces very rapidly). These values were based largely on experiments performed in the late 1950s involving heated irradiated UO_2 pellets. TID-14844 also included 1% of the remaining solid fission products, but these were dropped from consideration in Regulatory Guides 1.3 and 1.4. The 1% of the solid fission products are considered in certain areas such as equipment qualification.

Regulatory Guides 1.3 and 1.4 (Refs. 2 and 3) specify that the source term within containment is assumed to be instantaneously available for release and that the iodine chemical form is assumed to be predominantly (91%) in elemental (I_2) form, with 5% assumed to be particulate iodine and 4% assumed to be in organic form. These assumptions have significantly affected the design of engineered safety features. Containment isolation valve closure times have also been affected by these assumptions.

Use of the TID-14844 release has not been confined to an evaluation of site suitability and plant mitigation features such as sprays and filtration systems. The regulatory applications of this release are wide, including the basis for (1) the post-accident radiation environment for which safety-related equipment should be qualified, (2) post-accident habitability requirements for the control room, and (3) post-accident sampling systems and accessibility.

In contrast to the TID-14844 source term and containment leakage release used for design basis accidents, severe accident releases to the environment first arose in probabilistic risk assessments (e.g., Reactor Safety Study, WASH-1400 (Ref. 5)) in examining accident sequences that involved core melt and containments that could fail. Severe accident releases represent mechanistically determined best estimate releases to the environment, including estimates of failures of containment integrity. This is very different from the combination of the non-mechanistic release to containment postulated by TID-14844 coupled with the assumption of very limited containment leakage used for Part 100 siting calculations for design basis accidents. The worst severe accident releases resulting from containment failure or containment bypass can lead to consequences that are much greater than those associated with a TID-14844 source term released into containment where the containment is assumed to be leaking at its maximum leak rate for its design conditions. Indeed, some of the most severe releases arise from some containment bypass events, such as rupture of multiple steam generator tubes.

Although severe accident source terms have not been used in individual plant licensing safety evaluations, they have had significant regulatory applications. Source terms from severe accidents (beyond-design-basis accidents) came into regulatory consideration and usage shortly after the issuance of WASH-1400 in 1975, and their application was accelerated after the Three Mile Island accident in March 1979. Current applications rely to a large extent on the results of WASH-1400 and include (1) part of the basis for the sizes of emergency planning zones for all plants, (2) the basis for staff assessments of severe accident risk in

plant environmental impact statements, and (3) part of the basis for staff prioritization and resolution of generic safety issues, unresolved safety issues, and other regulatory analyses. Source term assessments based on WASH-1400 methodology appear in many probabilistic risk assessment studies performed to date.

1.2 Research Insights Since TID-14844

Source term estimates under severe accident conditions became of great interest shortly after the Three-Mile Island (TMI) accident when it was observed that only relatively small amounts of iodine were released to the environment compared with the amount predicted to be released in licensing calculations. This led a number of observers to claim that severe accident releases were much lower than previously estimated.

The NRC began a major research effort about 1981 to obtain a better understanding of fission-product transport and release mechanisms in LWRs under severe accident conditions. This research effort has included extensive NRC staff and contractor efforts involving a number of national laboratories as well as nuclear industry groups. These cooperative research activities resulted in the development and application of a group of computer codes known as the Source Term Code Package (STCP) (Ref. 6) to examine core-melt progression and fission product release and transport in LWRs. The NRC staff has also sponsored significant review efforts by peer reviewers, foreign partners in NRC research programs, industry groups, and the general public. The STCP methodology for severe accident source terms has also been reflected in NUREG-1150 (Ref. 7), which provides an updated risk assessment for five U.S. nuclear power plants.

As a result of the NRC's research effort to obtain a better understanding of fission product transport and release mechanisms in LWRs under severe accident conditions, the STCP emerged as an integral tool for analysis of fission product transport in the reactor coolant system (RCS) and containment. The STCP models release from the fuel with CORSOR (Ref. 8) and fission product retention and transport in the RCS with TRAPMELT (Ref. 9). Releases from core-concrete interactions are modeled using the VANESA and CORCON (Ref. 10) codes. Depending upon the containment type, SPARC or ICEDF (Refs. 11,12) are used in conjunction with NAUA (Ref.13) to model the transport and retention of fission product releases from the RCS and from core-concrete interactions into the containment, with subsequent release of fission products to the environment consistent with the state of the containment.

Improved modeling of severe accident phenomena, including fission product transport, has been provided by the recently developed MELCOR (Ref. 14) code. At this time, however, an insufficient body of calculations is available to provide detailed insights from this model.

Using analyses based on the STCP and MELCOR codes and NUREG-1150, the NRC has sponsored studies (Refs. 15-17) that analyzed the timing, magnitude, and duration of fission product releases. In addition, an examination and assessment of the chemical form of iodine likely to be found within containment as a result of a severe accident has also been carried out (Ref. 18).

In contrast to the instantaneous releases that were postulated in Regulatory Guides 1.3 and 1.4, analyses of severe accident sequences have shown that, despite differences in plant design and accident sequence, such releases can be generally categorized in terms of phenomenological phases associated with the degree of fuel melting and relocation, reactor pressure vessel integrity, and, as applicable, attack upon concrete below the reactor cavity by molten core materials. The general phases, or progression, of a severe LWR accident are shown in Table 1.1.

Table 1.1 Release Phases of a Severe Accident

Release Phases
Coolant Activity Release
Gap Activity Release
Early In-Vessel Release
Ex-Vessel Release
Late In-Vessel Release

Initially there is a release of coolant activity associated with a break or leak in the reactor coolant system. Assuming that the coolant loss cannot be accommodated by the reactor coolant makeup systems or the emergency core cooling systems, fuel cladding failure would occur with a release of the activity located in the gap between the fuel pellet and the fuel cladding.

As the accident progresses, fuel degradation begins, resulting in a loss of fuel geometry accompanied by gradual melting and slumping of core materials to the bottom of the reactor pressure vessel. During this period, the early in-vessel release phase, virtually all the noble gases and significant fractions of the volatile nuclides such as iodine and cesium are released into containment. The amounts of volatile nuclides released into containment during the early in-vessel phase are strongly influenced by the residence time of the radioactive material within the RCS during core degradation. High pressure sequences result in long

residence times and significant retention and plateout of volatile nuclides within the RCS, while low pressure sequences result in relatively short residence times and little retention within the RCS and consequently higher releases into containment.

If failure of the bottom head of the reactor pressure vessel occurs, two additional release phases may occur. Molten core debris released from the reactor pressure vessel into the containment will interact with the concrete structural materials of the cavity below the reactor (ex-vessel release phase). As a result of these interactions, quantities of the less volatile nuclides may be released into containment. Ex-vessel releases are influenced somewhat by the type of concrete in the reactor cavity. Limestone concrete decomposes to produce greater quantities of CO and CO_2 gases than basaltic concrete. These gases may, in turn, sparge some of the less volatile nuclides, such as barium and strontium, and small fractions of the lanthanides into the containment atmosphere. Large quantities of non-radioactive aerosols may also be released as a result of core-concrete interactions. The presence of water in the reactor cavity overlying any core debris can significantly reduce the ex-vessel releases (both radioactive and non-radioactive) into the containment, either by cooling the core debris, or at least by scrubbing the releases and retaining a large fraction in the water. The degree of scrubbing will depend, of course, upon the depth and temperature of any water overlying the core debris. Simultaneously, and generally with a longer duration, late in-vessel releases of some of the volatile nuclides, which had deposited in the reactor coolant system during the in-vessel phase, will also occur and be released into containment.

Two other phenomena that affect the release of fission products into containment could also occur, as discussed in Reference 7. The first of these is referred to as "high pressure melt ejection" (HPME). If the RCS is at high pressure at the time of failure of the bottom head of the reactor pressure vessel, quantities of molten core materials could be injected into the containment at high velocities. In addition to a potentially rapid rise in containment temperature, a significant amount of radioactive material could also be added to the containment atmosphere, primarily in the form of aerosols. The occurrence of HPME is precluded at low RCS pressures. A second phenomenon that could affect the release of fission products into containment is a possible steam explosion as a result of interactions between molten core debris and water. This could lead to fine fragmentation of some portion of the molten core debris with an increase in the amount of airborne fission products. While small scale steam explosions are considered quite likely to occur, they will not result in significant increases in the airborne activity already within containment. Large scale steam explosions, on the other hand, could result in significant increases in airborne activity, but are much less likely to occur. In any event, releases of particulates or vapors during steam explosions will also be accompanied by large amounts of water droplets, which would tend to quickly sweep released material from the atmosphere.

2 OBJECTIVES AND SCOPE

2.1 General

The primary objective of this report is to define a revised accident source term for regulatory application for future LWRs. The intent is to capture the major relevant insights available from recent severe accident research on the phenomenology of fission product release and transport behavior. The revised source term is expressed in terms of times and rates of appearance of radioactive fission products into the containment, the types and quantities of the species released, and other important attributes such as the chemical forms of iodine. This mechanistic approach will therefore present, for regulatory purposes, a more realistic portrayal of the amount of fission products present in the containment from a postulated severe accident.

2.2 Accidents Considered

In order to determine accident source terms for regulatory purposes, a range of severe accidents that have been analyzed for LWR plants was examined. Evaluation of a range of severe accident sequences was based upon work done in support of NUREG-1150 (Ref. 7). This work is documented in NUREG/CR-5747 (Ref. 17) and employed the integrated Source Term Code Package (STCP) computer codes, together with insights from the MELCOR code, which were used to analyze specific accident sequences of interest to provide the accident chronology as well as detailed estimates of fission product behavior within the reactor coolant system and the other pertinent parts of the plant. The sequences studied progressed to a complete core melt, involving failure of the reactor pressure vessel and including core-concrete interactions, as well.

A key decision to be made in defining an accident source term is the severity of the accident or group of accidents to be considered. Footnote 1 to 10 CFR Part 100 (Ref. 1), in referring to the postulated fission product release to be used for evaluating sites, notes that "Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products." Possible choices range from (1) slight fuel damage accidents involving releases into containment

of a small fraction of the volatile nuclides such as the noble gases, (2) severe core damage accidents involving major fuel damage but without reactor vessel failure or core-concrete interactions (similar in severity to the TMI accident), or (3) complete core-melt events with core-concrete interactions. These outcomes are not equally probable. Since many reactor systems must fail for core degradation with reactor vessel failure to occur and core-concrete interactions to occur, one or more systems may be returned to an operable status before core melt commences. Hence, past operational and accident experience together with information on modern plant designs, together with a vigorous program aimed at developing accident management procedures, indicate that complete core-melt events resulting in reactor pressure vessel failure are considerably less likely to occur than those involving major fuel damage without reactor pressure vessel failure, and that these, in turn, are less likely to occur than those involving slight fuel damage.

For completeness, this report displays the mean or average release fractions for all the release phases associated with a complete core melt. However, it is concluded that any source term selected for a particular regulatory application should appropriately reflect the likelihood associated with its occurrence.

It is important to emphasize that the release fractions for the source terms presented in this report are intended to be representative or typical, rather than conservative or bounding values, of those associated with a low pressure core-melt accident, except for the initial appearance of fission products from failed fuel, which was chosen conservatively. The release fractions are not intended to envelope all potential severe accident sequences, nor to represent any single sequence, since accident sequences yielding both higher as well as lower release fractions were examined and factored into the final report presented here.

Source terms for future reactors may differ from those presented in this report which are based upon insights derived from current generation light-water reactors. An applicant may propose changes in source term parameters (timing, release magnitude, and chemical form) from those contained in this report, based upon and justified by design specific features.

The NRC staff also intends to allow credit for removal or reduction of fission products within containment via engineered features provided for fission product reduction such as sprays or filters, as well as by natural processes such as aerosol deposition. These are discussed in Section 5.

2.3 Limitations

The accident source terms defined in this report have been derived from examination of a set of severe accident sequences for LWRs of current design. Because of general similarities in plant and core design parameters, these results are also considered to be applicable to evolutionary LWR designs such as General Electric's Advanced Boiling Water Reactor (ABWR) and Combustion Engineering's (CE) System 80+.

Currently, the NRC staff is reviewing reactor designs for several smaller LWRs employing some passive features for core cooling and containment heat removal. While the "passive" plants are generally similar to present LWRs, they are expected to have somewhat lower core power densities than those of current LWRs. Hence, an accident for the passive plants similar to those used in this study would likely extend over a longer time span. For this reason, the timing and duration values provided in the release tables given in Section 3.3 are probably shorter than those applicable to the passive plants. The release fractions shown may also be overestimated somewhat for high pressure sequences associated with the passive plants, since longer times for accident progression would also allow for enhanced retention of fission products in the primary coolant system during core heatup and degradation. Despite the lack of specific accident sequence information for these designs, the in-containment accident source terms provided below may be considered generally applicable to the "passive" designs.

The accident source terms provided in this report are not considered applicable to reactor designs that are very different from LWRs, such as high-temperature gas-cooled reactors or liquid-metal reactors.

Recent information has indicated that high burnup fuel, that is, fuel irradiated at levels in excess of about 40 GWD/MTU, may be more prone to failure during design basis reactivity insertion accidents (RIA) than previously thought. Preliminary indications are that high burnup fuel also may be in a highly fragmented or powdered form, so that failure of the cladding could result in a significant fraction of the fuel itself being released. In contrast, the source term contained in this report is based upon fuel behavior results obtained at lower burnup levels where the fuel pellet remains intact upon cladding failure, resulting in a release only of those fission product gases residing in the gap between the fuel pellet and the cladding. Because of this recent information regarding high burnup fuels, the NRC staff cautions that, until further information indicates otherwise, the source term in this report (particularly gap activity) may not be applicable for fuel

irradiated to high burnup levels (in excess of about 40 GWD/MTU).

3 ACCIDENT SOURCE TERMS

The expression "in-containment source terms," as used in this report, denotes the fission product inventory present in the containment at any given time during an accident. To evaluate the in-containment source term during the course of an accident, the time-history of the fission product release from the core into the containment must be known, as well as the effect of fission product removal mechanisms, both natural and engineered, to remove radioactive materials from the containment atmosphere. This section discusses the time-history of the fission product releases into the containment. Removal mechanisms are discussed in Section 5.

3.1 Accident Sequences Reviewed

All the accident sequences identified in NUREG-1150 were reviewed and some additional Source Term Code Package (STCP) and MELCOR calculations were performed. The dominant sequences which are considered to significantly impact the source term are summarized in Table 3.1 for BWRs and Table 3.2 for PWRs.

3.2 Onset of Fission Product Release

This section discusses the assumptions used in selecting the scenario appropriate for defining the early phases of the source term (coolant activity and gap release phases). It was considered appropriate to base these early release phases on the design basis initiation that could lead to earliest fuel failures.

A review of current plant final safety analysis reports (FSARs) was made to identify all design basis accidents in which the licensee had identified fuel failure. For all accidents with the potential for release of radioactivity into the environment, the class of accident that had the shortest time until the first fuel rod failed was the design basis LOCA. As might be expected, the time until cladding failure is very sensitive to the design of the reactor, the type of accident assumed, and the fuel rod design. In particular, the maximum linear heat generation rate, the internal fuel rod pressure, and the stored energy in the fuel rod are significant considerations.

Table 3.1 BWR Source Term Contributing Sequences

Plant	Sequence	Description
Peach Bottom	TC1	ATWS with reactor depressurized
	TC2	ATWS with reactor pressurized
	TC3	TC2 with wetwell venting
	TB1	SBO with battery depletion
	TB2	TB1 with containment failure at vessel failure
	S2E1	LOCA (2"), no ECCS and no ADS
	S2E2	S2E1 with basaltic concrete
	V	RHR pipe failure outside containment
	TBUX	SBO with loss of all DC power
LaSalle	TB	SBO with late containment failure
Grand Gulf	TC	ATWS early containment failure fails ECCS
	TB1	SBO with battery depletion
	TB2	TB1 with H_2 burn fails containment
	TBS	SBO, no ECCS but reactor depressurized
	TBR	TBS with AC recovery after vessel failure

SBO	Station Blackout	LOCA	Loss of Coolant Accident
RCP	Reactor Coolant Pump	RHR	Residual Heat Removal
ADS	Automatic Depressurization System	ATWS	Anticipated Transient Without Scram

Table 3.2 PWR Source Term Contributing Sequences

Plant	Sequence	Description
Surry	AG	LOCA (hot leg), no containment heat removal systems
	TMLB'	LOOP, no PCS and no AFWS
	V	Interfacing system LOCA
	S3B	SBO with RCP seal LOCA
	S2D-δ	SBLOCA, no ECCS and H_2 combustion
	S2D-β	SBLOCA with 6" hole in containment
Zion	S2DCR	LOCA (2"), no ECCS no CSRS
	S2DCF1	LOCA RCP seal, no ECCS, no containment sprays, no coolers—H_2 burn or DCH fails containment
	S2DCF2	S2DCF1 except late H_2 or overpressure failure of containment
	TMLU	Transient, no PCS, no ECCS, no AFWS—DCH fails containment
Oconee 3	TMLB'	SBO, no active ESF systems
	S1DCF	LOCA (3"), no ESF systems
Sequoyah	S3HF1	LOCA RCP, no ECCS, no CSRS with reactor cavity flooded
	S3HF2	S3HF1 with hot leg induced LOCA
	3HF3	S3HF1 with dry reactor cavity
	S3B	LOCA (½") with SBO
	TBA	SBO induces hot leg LOCA—hydrogen burn fails containment
	ACD	LOCA (hot leg), no ECCS no CS
	S3B1	SBO delayed 4 RCP seal failures, only steam driven AFW operates
	S3HF	LOCA (RCP seal), no ECCS, no CSRS
	S3H	LOCA (RCP seal) no ECC recirculation

SBO	Station Blackout	LOCA	Loss of Coolant Accident
RCP	Reactor Coolant Pump	DCH	Direct Containment Heating
PCS	Power Conversion System	ESF	Engineered Safety Feature
CS	Containment Spray	CSRS	CS Recirculation System
ATWS	Anticipated Transient Without Scram	LOOP	Loss of Offsite Power

The details of the specific accident sequences are documented in NUREG/CR-5747, Estimate of Radionuclide Release Characteristics into Containment Under Severe Accident Conditions (Ref. 17).

To determine whether a design basis LOCA was a reasonable scenario upon which to base the timing of initial fission product release into the containment, various PRAs were reviewed to determine the contribution to core damage frequency (CDF) resulting from LOCAs. This information is shown in Table 3.3. As can be seen from this table, LOCAs are a small contributor to CDF for BWRs, but can be a substantial contributor for PWRs. Therefore, for PWRs a large LOCA is considered a reasonable initiator to assume for modeling the earliest appearance of the gap activity if the plant has not been approved for leak before break (LBB) operation. For plants that have received LBB approval, a small LOCA (6" line break) would more appropriately model the timing. For BWRs, large LOCAs may not be an appropriate scenario for gap activity timing. However, since the time to initial fuel rod failure is long for BWRs, even for large LOCAs,

use of the large LOCA scenario should not unduly penalize BWRs and will maintain consistency with the assumptions for the PWR. As with the PWR, for an LBB approved plant, the timing associated with a small LOCA (6" line break) would be more appropriate.

In order to provide a realistic estimate of the shortest time for fuel rod failure for the LOCA, calculations were performed using the FRAPCON2, SCDAP/RELAP5 MOD 3.0, and FRAPT6 computer codes for two plants. The two plants were a Babcock and Wilcox (B&W) plant with a 15 by 15 fuel rod array and a Westinghouse 4-loop (W) plant with a 17 by 17 fuel rod array. For each plant, a sensitivity study was performed to identify the size of the LOCA that resulted in the shortest fuel rod failure time (Ref. 15). In both cases, the accident was a double-ended guillotine rupture of the cold leg pipe. The minimum time from the time of accident initiation until first fuel rod failure was calculated to be 13 and 24.6 seconds for the B&W and W plants, respectively. A sensitivity study was performed to determine the effect of tripping or not tripping the reactor coolant pumps. The results indicated that tripping of the reactor coolant pumps had no appreciable impact on timing. For a 6-inch line break, the time until first fuel rod failure is expected to be greater than 6.5 and 10 minutes, respectively.

Table 3.3 Contribution of LOCAs to Core Damage Frequency (CDF)—Internal Events

Boiling Water Reactors	Percent of CDF caused by LOCAs	Percent of CDF caused by large LOCAs (>6" line break)
Peach Bottom (NUREG-1150)	3.5	1.0
Grand Gulf (NUREG-1150)	0.1	0.03
Millstone 1 (Utility)	23	13
Pressurized Water Reactors		
Surry (NUREG-1150)	15	4.3
Sequoyah (NUREG-1150)	63	4.6
Zion (NUREG-1150)	87	1.4
Calvert Cliffs (IREP)	21	<1
Oconee-3 (EPRI/NSAC)	43	3.0

A comparison calculation was done using the TRAC-PF1 MOD 1 code, version 14.3U5Q.LG on the W plant. This analysis indicated that the first fuel rod failure would occur 34.9 seconds after pipe rupture, in contrast to the value of 24.6 seconds calculated using SCDAP/RELAP. The reasons for the difference between the SCDAP/RELAP5 MOD 3.0 and TRAC-PF1 MOD 1 are discussed in Reference 15.

The review of the FSARs for BWRs indicates that fuel failures may occur significantly later, on the order of several minutes or more. No calculations have been performed using the aforementioned suite of codes.

For determining the time of appearance of gap activity in the containment (i.e., initial fuel failure), which corresponds to the duration of the coolant activity phase and the beginning of the gap activity phase, it would be appropriate to perform a plant specific calculation using the codes described above. However, if no plant specific calculations are performed, the minimum times discussed above may be used to provide an estimate of the earliest time to fuel rod failure.

Source terms for future reactors may differ from those presented in this report which are based upon insights derived from current generation light-water reactors. An applicant may propose changes in source term parameters (timing, release magnitude, and chemical form) from those contained in this report, based upon and justified by design specific features.

3.3 Duration of Release Phases

Section 1.2 provided a qualitative discussion of the release phases of an accident. This section provides estimated durations for these release phases.

The coolant activity phase begins with a postulated pipe rupture and ends when the first fuel rod has been estimated to fail. During this phase, the activity released to the containment atmosphere is that associated with very small amounts of radioactivity dissolved in the coolant itself. As discussed in Section 3.2 above, this phase is estimated to last about 25 seconds for Westinghouse PWRs, and about 13 seconds for B&W PWRs, assuming a large break LOCA. For a smaller LOCA (e.g., a 6-inch line break), such as would

be considered for a plant that has received LBB approval, the coolant activity phase duration would be expected to be at least 10 minutes. Although not specifically evaluated at this time, Combustion Engineering (CE) PWRs would be expected to have coolant activity durations similar to Westinghouse plants. For BWRs, the coolant activity phase would be expected to last longer; however, unless plant specific calculations are made, the durations discussed above are considered applicable.

The gap activity release phase begins when fuel cladding failure commences. This phase involves the release of that radioactivity that has collected in the gap between the fuel pellet and cladding. This process releases to containment a few percent of the total inventory of the more volatile radionuclides, particularly noble gases, iodine, and cesium. During this phase, the bulk of the fission products continue to be retained in the fuel itself. The gap activity phase ends when the fuel pellet bulk temperature has been raised sufficiently that significant amounts of fission products can no longer be retained in the fuel. As noted in Reference 16, a review of STCP calculated results for six reference plants, PWRs as well as BWRs, indicated that significant fission product releases from the bulk of the fuel itself were estimated to commence no earlier than about 30 minutes and 60 minutes for PWRs and BWRs, respectively, after the onset of the accident. However, more recent calculations (Ref. 19) for the Peach Bottom plant using the MELCOR code indicated that the durations of the gap release for three BWR accident sequences were about 30 minutes, as well. On this basis, the duration of the gap activity release phase has been selected to be 0.5 hours, for both PWRs and BWRs.

During the early in-vessel release phase, the fuel as well as other structural materials in the core reach sufficiently high temperatures that the reactor core geometry is no longer maintained and fuel and other materials melt and relocate to the bottom of the reactor pressure vessel. During this phase, significant quantities of the volatile nuclides in the core inventory as well as small fractions of the less volatile nuclides are estimated to be released into containment. This release phase ends when the bottom head of the reactor pressure vessel fails, allowing molten core debris to fall onto the concrete below the reactor pressure vessel. Release durations for this phase vary depending on both the reactor type and the accident sequence. Tables 3.4 and 3.5, based on results from Reference 16, show the estimated duration times for PWRs and BWRs, respectively.

Table 3.4 In-Vessel Release Duration for PWR Sequences

Plant	Accident Sequence*		Release Duration (Min)
Surry	TMLB'	(H)	41
Surry	S3B	(H)	36
Surry	AG	(L)	215
Surry	V	(L)	104
Zion	TMLU	(H)	41
Zion	S2DCR/S2DCF	(H)	39
Sequoyah	S3HF/S3B	(H)	46
Sequoyah	S3B1	(H)	75
Sequoyah	TMLB'	(H)	37
Sequoyah	TBA	(L)	195
Sequoyah	ACD	(L)	73
Oconee	TMLB'	(H)	35
Oconee	S1DCF	(L)	84

*(H or L) Denotes whether the accident occurs at high or low pressure.

Based on the information in these tables, the staff concludes that the in-vessel release phase is somewhat longer for BWR plants than for PWR plants. This is largely due to the lower core power density in BWR plants that extends the time for complete core melt. Representative times for the duration of the in-vessel release phase have been selected to be 1.3 hours and 1.5 hours, for PWR and BWR plants respectively, as recommended by Reference 17.

The ex-vessel release phase begins when molten core debris exits the reactor pressure vessel and ends when

Table 3.5 In-Vessel Release Duration for BWR Sequences

Plant	Accident Sequence*		Release Duration (Min)
Peach Bottom	TC2	(H)	66
Peach Bottom	TC3		68
Peach Bottom	TC1	(L)	97
Peach Bottom	TB1/TB2	(H)	91
Peach Bottom	V	(L)	69
Peach Bottom	S2E	(H)	81
Peach Bottom	TBUX	(H)	67
LaSalle	TB	(H)	81
Grand Gulf	TB	(H)	122
Grand Gulf	TC1	(L)	130
Grand Gulf	TBS/TBR	(L)	96

*(H or L) denotes whether the accident occurs at high or low pressure.

the debris has cooled sufficiently that significant quantities of fission products are no longer being released. During this phase, significant quantities of the volatile radionuclides not already released during the early in-vessel phase as well as lesser quantities of non-volatile radionuclides are released into containment. Although releases from core-concrete interactions are predicted to take place over a number of hours after vessel breach, Reference 16 indicates that the bulk of the fission products (about 90%), with the exception of tellurium and ruthenium, are expected to be released over a 2-hour period for PWRs and a 3-hour period for BWRs. For tellurium and ruthenium, ex-vessel releases extend over 5 and 6 hours, respectively, for PWRs and BWRs. The difference in duration of the ex-vessel phase between PWRs and BWRs is largely attributable to the larger amount of zirconium in BWRs, which provides additional chemical energy of oxidation. Based on Reference 17, the ex-vessel release phase duration is taken to be 2 and 3 hours, respectively, for PWRs and BWRs.

The late in-vessel release phase commences at vessel breach and proceeds simultaneously with the occurrence of the ex-vessel phase. However, the duration is not the same for both phases. During this release phase, some of the volatile nuclides deposited within the reactor coolant system earlier during core degradation and melting may re-volatilize and be released into containment. Reference 17, after a review of the source term uncertainty methodology used in NUREG-1150 (Ref. 7), estimates the late in-vessel release phase to have a duration of 10 hours. This value has been selected for this report.

A summary of the release phases and the selected duration times for PWRs and BWRs is shown for reference purposes in Table 3.6.

Table 3.6
Release Phase Durations for PWRs and BWRs

Release Phase	Duration, PWRs (Hours)	Duration, BWRs (Hours)
Coolant Activity	10 to 30 seconds*	30 seconds*
Gap Activity	0.5	0.5
Early In-Vessel	1.3	1.5
Ex-Vessel	2	3
Late In-Vessel	10	10

*Without approval for leak-before-break. Coolant activity phase duration is assumed to be 10 minutes with leak-before-break approval.

3.4 Fission Product Composition and Magnitude

In considering severe accidents in which the containment might fail, WASH-1400 (Ref. 5) examined the spectrum of fission products and grouped 54 radionuclides into 7 major groups on the basis of similarity in chemical behavior. The effort associated with the STCP

further analyzed these groupings and expanded the 7 fission product groups into 9 groups. These are shown in Table 3.7.

Table 3.7 STCP Radionuclide Groups

Group	Elements
1	Xe, Kr
2	I, Br
3	Cs, Rb
4	Te, Sb, Se
5	Sr
6	Ru, Rh, Pd, Mo, Tc
7	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y
8	Ce, Pu, Np
9	Ba

Both the results of the STCP analyses and the uncertainty analysis (using the results of the NUREG-1150 source term expert panel elicitation) reported in NUREG/CR-5747 (Ref. 17) indicate only minor differences between Ba and Sr releases. Hence, a revised grouping of radionuclides has been developed that groups Ba and Sr together. The relative importance to offsite health and economic consequences of the radioactive elements in a nuclear reactor core has been examined and documented in NUREG/CR-4467 (Ref. 20). In addition to the elements already included in Table 3.7, Reference 20 found that other elements such as Curium could be important for radiological consequences if released in sufficiently large quantities. For this reason, group 7 has been revised to include Curium (Cm) and Americium (Am), while group 6 has been revised to include Cobalt (Co). The revised radionuclide groups used in this report including revised titles and the elements comprising each group are shown in Table 3.8.

Source term releases into the containment were evaluated by reactor type, i.e., BWR or PWR, from the sequences in NUREG-1150 and the supplemental STCP calculations discussed in Section 3.1.

Releases into containment during the early in-vessel phase, prior to reactor pressure vessel failure, are markedly affected by retention in the RCS, which is a function of the residence time in the RCS during core degradation. High pressure in the RCS during core degradation allows for longer residence time of aerosols released from the core. This, in turn, permits increased retention of aerosols within the RCS and lower releases from the core into the containment.

Similarly, low pressure sequences cause aerosols generated within the RCS to be swept out rapidly without significant retention within the RCS, thereby resulting in higher release fractions from the core into containment.

Table 3.8 Revised Radionuclide Groups

Group	Title	Elements in Group
1	Noble gases	Xe, Kr
2	Halogens	I, Br
3	Alkali Metals	Cs, Rb
4	Tellurium group	Te, Sb, Se
5	Barium, strontium	Ba, Sr
6	Noble Metals	Ru, Rh, Pd, Mo, Tc, Co
7	Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
8	Cerium group	Ce, Pu, Np

The relative frequency of occurrence of high vs. low pressure sequences were examined for both BWRs and PWRs. The results of this survey are shown in Table 3.9, and they indicate that a significant fraction of the sequences examined, in terms of frequency, occurred at low pressure. In addition, advanced PWR designs are increasingly incorporating safety-grade depressurization systems, primarily to minimize the likelihood of high pressure melt ejection (HPME) with its associated high containment atmosphere heat loads and large amounts of atmospheric aerosols.

For these reasons, the composition and magnitude of the source term has been chosen to be representative of conditions associated with low pressure in the RCS at the time of reactor core degradation and pressure vessel failure. Reference 17 provides estimates of the mean core fractions released into containment, as estimated by NUREG-1150 (Ref. 7), for accident sequences occurring under low RCS pressure and high zirconium oxidation conditions. These are shown in Tables 3.10 and 3.11.

3.5 Chemical Form

The chemical form of iodine and its subsequent behavior after entering containment from the reactor coolant system have been documented in NUREG/CR-5732, Iodine Chemical Forms in LWR Severe Accidents (Ref. 18) and in ORNL/TM-12202, "Models of Iodine Behavior in Reactor Containments," (Ref. 21).

Table 3.9 Fraction of mean core damage frequency with high, intermediate, and low pressure sequences (internal events only unless otherwise noted)

Boiling Water Reactors	High Pressure at Vessel Breach	Intermed. press. at vessel breach	Low Pressure at Vessel Breach (<200 psi)	No vessel breach
LaSalle— external events only	0.27	N/A	0.67	0.06
LaSalle— internal events only	0.19	N/A	0.62	0.19
Grand Gulf	0.28	N/A	0.51	0.21
Peach Bottom	0.51	N/A	0.41	0.08
Pressurized Water Reactors				
Surry	0.06	0.07	0.37	0.50
Sequoyah	0.14	0.21	0.24	0.41
Zion	0.03	0.15	0.72	0.10

Table 3.10 Mean Values of Radionuclides Into Containment for BWRs, Low RCS Pressure, High Zirconium Oxidation

Nuclide	Early In-Vessel	Ex-Vessel	Late In-vessel
N. G.	1.0	0	0
I	0.27	0.37	0.07
Cs	0.2	0.45	0.03
Te	0.11	0.38	0.01
Sr	0.03	0.24	0
Ba	0.03	0.21	0
Ru	0.007	0.004	0
La	0.002	0.01	0
Ce	0.009	0.01	0

Table 3.11 Mean Values of Radionuclide Releases Into Containment for PWRs, Low RCS Pressure, High Zirconium Oxidation

Nuclide	Early In-Vessel	Ex-Vessel	Late In-vessel
N.G.	1.0	0	0
I	0.4	0.29	0.07
Cs	0.3	0.39	0.06
Te	0.15	0.29	0.025
Sr	0.03	0.12	0
Ba	0.04	0.1	0
Ru	0.008	0.004	0
La	0.002	0.015	0
Ce	0.01	0.02	0

The results from Ref. 18 indicate that iodine entering the containment is at least 95% CsI with the remaining 5% as I plus HI, with not less than 1% of each as I and HI. Once the iodine enters containment, however, additional reactions are likely to occur. In an aqueous environment, as expected for LWRs, iodine is expected to dissolve in water pools or plate out on wet surfaces in ionic form as I^-. Subsequently, iodine behavior within containment depends on the time and pH of the water solutions. Because of the presence of other dissolved fission products, radiolysis is expected to occur and lower the pH of the water pools. Without any pH control, the results indicate that large fractions of the dissolved iodine will be converted to elemental iodine and be released to the containment atmosphere. However, if the pH is controlled and maintained at a value of 7 or greater, very little (less than 1%) of the dissolved iodine will be converted to elemental iodine. Some considerations in achieving pH control are discussed in NUREG/CR-5950, "Iodine Evolution and pH Control," (Ref. 22).

Organic compounds of iodine, such as methyl iodide, CH_3I, can also be produced over time largely as a result of elemental iodine reactions with organic materials. Organic iodide formation as a result of reactor accidents has been surveyed in WASH-1233, "Review of Organic Iodide Formation Under Accident Conditions in Water-Cooled Reactors," (Ref. 23), and more recently in NUREG/CR-4327, "Organic Iodide Formation Following Nuclear Reactor Accidents," (Ref.24). From an analysis of a number of containment experiments, WASH-1233 concluded that, considering both non-radiolytic as well as radiolytic means, no more than 3.2 percent of the airborne iodine would be converted to organic iodides during the first two hours following a fission product release. The value of 3.2 percent was noted as a conservative upper limit and was judged to be considerably less, since it did not account, among other things, for decreased radiolytic formation of organic iodide due to iodine removal mechanisms within containment. Reference 24 also included results involving irradiated fuel elements, and concluded that the organic iodide concentration within containment would be about 1 percent of the iodine release concentration over a wide range of iodine concentrations.

A conversion of 4 percent of the elemental iodine to organic has been implicitly assumed by the NRC staff in Regulatory Guides 1.3 and 1.4, based upon an upper bound evaluation of the results in WASH-1233. However, in view of the results of Ref. 23 that a conversion of 3.2 percent is unduly conservative, a value of 3 percent is considered more realistic and will be used in this report. Where the pH is controlled at values of 7 or greater within the containment, elemental iodine can be taken as comprising no more than 5 percent of the total iodine released, and iodine in organic form may be taken as comprising no greater than 0.15 percent (3 percent of 5 percent) of the total iodine released.

Organic iodide formation in BWRs versus PWRs is not notably different. Reference 18 examined not only iodine entering containment as CsI; but also considered other reactions that might lead to volatile forms of iodine within containment, such as reactions of CsOH with surfaces and revaporization of CsI from RCS surfaces. Reference 18 indicates (Table 2.4) that for the Peach Bottom TC2 sequence, the estimated percentage of iodine as HI was 3.2 percent, not notably less than the PWR sequences examined. While organic iodide is formed largely from reactions of elemental iodine, Ref. 22 clearly notes that reactions with HI may be important.

Although organic iodine is not readily removed by containment sprays or filter systems, it is unduly conservative to assume that organic iodine is not removed at all from the containment atmosphere, once generated, since such an assumption can result in an overestimate of long-term doses to the thyroid. References 23 and 24 discuss the radiolytic destruction of organic iodide, and Standard Review Plan Section (S.R.P.) 6.5.2 notes the above reference and indicates that removal of organic iodide may be considered on a case-by-case basis. A rational model for organic iodine behavior within containment would consider both its formation as well as destruction in a time-dependent fashion. Development of such a model, however, is beyond the scope of the present report.

Clearly, where the pH is not controlled to values of 7 or greater, significantly larger fractions of elemental iodine, as well as organic iodine may be expected within containment.

All other fission products, except for the noble gases and iodine, discussed above, are expected to be in particulate form.

3.6 Proposed Accident Source Terms

The proposed accident source terms, including their timing as well as duration, are listed in Tables 3.12 for BWRs and 3.13 for PWRs. The information for these tables was derived from the simplification of the NUREG-1150 (Ref. 7) source terms documented in NUREG/CR-5747 (Ref. 17). It should also be noted that the rate of release of fission products into the containment is assumed to be constant during the duration time shown.

Table 3.12 BWR Releases Into Containment*

	Gap Release***	Early In-Vessel	Ex-Vessel	Late In-Vessel
Duration (Hours)	0.5	1.5	3.0	10.0
Noble Gases**	0.05	0.95	0	0
Halogens	0.05	0.25	0.30	0.01
Alkali Metals	0.05	0.20	0.35	0.01
Tellurium group	0	0.05	0.25	0.005
Barium, Strontium	0	0.02	0.1	0
Noble Metals	0	0.0025	0.0025	0
Cerium group	0	0.0005	0.005	0
Lanthanides	0	0.0002	0.005	0

* Values shown are fractions of core inventory.
** See Table 3.8 for a listing of the elements in each group
*** Gap release is 3 percent if long-term fuel cooling is maintained.

Table 3.13 PWR Releases Into Containment*

	Gap Release***	Early In-Vessel	Ex-Vessel	Late In-Vessel
Duration (Hours)	0.5	1.3	2.0	10.0
Noble Gases**	0.05	0.95	0	0
Halogens	0.05	0.35	0.25	0.1
Alkali Metals	0.05	0.25	0.35	0.1
Tellurium group	0	0.05	0.25	0.005
Barium, Strontium	0	0.02	0.1	0
Noble Metals	0	0.0025	0.0025	0
Cerium group	0	0.0005	0.005	0
Lanthanides	0	0.0002	0.005	0

* Values shown are fractions of core inventory.
** See Table 3.8 for a listing of the elements in each group
*** Gap release is 3 percent if long-term fuel cooling is maintained.

It is emphasized that the release fractions for the source terms presented in this report are intended to be representative or typical, rather than conservative or bounding values, of those associated with a low pressure core-melt accident, except for the initial appearance of fission products from failed fuel, which was chosen conservatively. The release fractions are not intended to envelope all potential severe accident sequences, nor to represent any single sequence.

Tables 3.12 and 3.13 in this, the final report, were modified from the tables in the draft report which were taken from Table 3.9 and Table 3.10, for BWRs and PWRs, respectively. The changes and the reasons for these was as follows:

1. BWR in-vessel release fractions for the volatile nuclides (I and Cs) increased slightly while ex-vessel release fractions for the same nuclides was reduced as a result of comments received and additional MELCOR calculations available after issuance of the draft report. The total I and Cs released into containment over all phases of the accident remained the same.

2. Release fractions for Te, Ba and Sr were reduced somewhat, both for in-vessel as well as ex-vessel releases, in response to comments.

3. Release fractions for the non-volatile nuclides, particularly during the early in-vessel phase were reduced significantly based on additional research results (Ref. 25) since issuance of NUREG-1150 which indicate that releases of low volatile nuclides, both in-vessel as well as ex-vessel, have been overestimated. A re-examination in response to comments received showed that the supposed "means" of the uncertainty distribution were in excess of other measures of the distribution, such as the 75th percentile. In this case, the 75th percentile was selected as an appropriate measure of the release fraction. For additional discussion on this topic, see Section 4.4.

4. Gap activity release fractions were reduced from 5 percent to 3 percent for accidents not involving degraded or molten core conditions, and where long-term fuel cooling is maintained. See additional discussion below.

Based on WASH-1400 (Ref. 5), the inventory of fission products residing in the gap between the fuel and the cladding is no greater than 3 percent except for cesium, which was estimated to be about 5 percent. NUREG/CR-4881 (Ref.16) reported a comparison of more recently available estimations and observations indicating that releases of the dominant fission product groups were generally below the values reported in Reference 5. However, the magnitude of fission products released during the gap release phase can vary, depending upon the type of accident. Accidents where fuel failures occur may be grouped as follows:

1. Accidents where long-term fuel cooling is maintained despite fuel failure. Examples include the design basis LOCA where ECCS functions, and a postulated spent fuel handling accident. For this category, fuel failure is taken to result in an immediate release, based upon References 5 and 16, of 3 percent of the volatile fission products (noble gases, iodine, and cesium) which are in the gap between the fuel pellet and the cladding. No subsequent appreciable release from the fuel pellet occurs, since the fuel does not experience prolonged high temperatures.

2. Accidents where long-term fuel cooling or core geometry are not maintained. Examples include degraded core or core-melt accidents, including the postulated limiting design basis fission product release into containment used to show compliance with 10 CFR Part 100. For this category, the gap release phase may overlap to some degree with the early in-vessel release phase. The release magnitude has been taken as an initial release of 3 percent of the volatiles (as for category 1), plus an additional release of 2 percent over the duration of the gap release phase.

3. Accidents where fuel failure results from reactivity insertion accidents (RIA), such as the postulated rod ejection (PWR) or rod drop (BWR) accidents. The accidents examined in this report do not contain information on reactivity induced accidents to permit a quantitative discussion of fission product releases from them. Hence, the gap release magnitude presented in Tables 3.12 and 3.13 may not be applicable to fission product releases resulting from reactivity insertion accidents.

Recent information has indicated that high burnup fuel, that is, fuel irradiated at levels in excess of about 40 GWD/MTU, may be more prone to failure during design basis reactivity insertion accidents than previously thought. Preliminary indications are that high burnup fuel also may be in a highly fragmented or powdered form, so that failure of the cladding could result in a significant fraction of the fuel itself being released. In contrast, the source term contained in this report is based upon fuel behavior results obtained at lower burnup levels where the fuel pellet remains intact upon cladding failure, resulting in a release only of those fission product gases residing in the gap between the fuel pellet and the cladding. Because of this recent information regarding high burnup fuels, the NRC staff cautions that, until further information indicates otherwise, the source term in Tables 3.12 and 3.13 (particularly gap activity) may not be applicable for fuel irradiated to high burnup levels (in excess of about 40 GWD/MTU).

With regard to the ex-vessel releases associated with core-concrete interactions, according to Reference 17, there were only slight differences in the fission products released into containment between limestone vs. basaltic concrete. Hence, the table shows the releases only for a limestone concrete. Further, the releases shown for the ex-vessel phase are assumed to be for a dry reactor cavity having no water overlying any core debris. Where water covers the core debris, aerosol scrubbing will take place and reduce the quantity of aerosols entering the containment atmosphere. See Section 5.4 for further information.

3.7 Nonradioactive Aerosols

In addition to the fission product releases into containment shown in Tables 3.12 and 3.13, quantities of nonradioactive or relatively low activity aerosols will also be released into containment. These aerosols arise from core structural and control rod materials released during the in-vessel phase and from concrete decomposition products during the ex-vessel phase. A detailed

analysis of the quantity of nonfission product aerosols released into containment was not undertaken. Precise estimates of the masses of non-radioactive aerosols released into containment are difficult to determine.

Reference 26 evaluated one PWR sequence (Sequoyah) and one BWR (Peach Bottom) sequence and calculated in-vessel non-radioactive aerosol masses of 350 and 780 kilograms, respectively, for the PWR and BWR sequences. The same reference calculated that ex-vessel aerosol masses (assuming a dry cavity) would be higher, 3800 and 5600 kilograms, respectively, for the PWR and BWR sequences investigated. However, these values, particularly for the ex-vessel release phase, may be excessive. NUREG/CR–4624 (Ref. 27) examined several sequences for both PWRs and BWRs and calculated ex-vessel releases to containment of about 1000 and 4000 kilograms, respectively, for PWRs and BWRs. NUREG/CR–5942 (Ref.19), making use of the MELCOR code, calculated significantly lower releases during the ex-vessel phase of about 1000 kilograms for the Peach Bottom plant.

In view of the wide diversity of calculated results, the NRC staff concludes that precise estimates of the release of non-radioactive aerosols are not available at this time. Because nonradioactive aerosol masses could have an effect upon the operation of certain plant equipment, such as filter loadings or sump performance, during and following an accident, however, the NRC staff concludes that the release of non-radioactive aerosols should be considered by the designer using methods considered applicable for his design, and the potential impact upon the plant evaluated.

4 MARGINS AND UNCERTAINTIES

This section discusses some of the more significant conservatisms and margins in the proposed accident source term given in Section 3. Briefly, the proposed release fractions have been developed from a complete core-melt accident, that is, assuming core melt with reactor pressure vessel failure and with the assumption of core-concrete interactions. The timing aspects were selected to be typical of a low pressure core-melt scenario, except that the onset of the release of gap activity was based upon the earliest calculated time of fuel rod failure under accident conditions. The magnitude of the fission products released into containment was intended to be representative and, except for the low volatile nuclides, as discussed in section 4.4, was estimated from the mean values for a typical low-pressure core-melt scenario.

4.1 Accident Severity and Type

As noted earlier in Section 2.2, this report discusses mean or average release fractions for all the release phases associated with a complete core-melt accident, including reactor pressure vessel failure. The accident selected is one in which core melt occurs at low pressure conditions. A low pressure core melt scenario results in a relatively low level of fission product retention within the reactor coolant system, and a consequently high level of release of fission products from the core into containment during the early in-vessel release phase. Since the bulk of the fission products entering containment do so during the early in-vessel release phase, selection of a low pressure core melt scenario provides a high estimate of the total quantity of fission products released into containment, as well as that during the early in-vessel release phase.

4.2 Onset of Fission Product Release

The onset, or earliest time of appearance of fission products within containment, has been selected on the basis of the earliest time to failure of a fuel rod, given a design basis LOCA. This is estimated to be from about 13 to 25 seconds for plants that do not have leak-before-break approval for their reactor coolant system piping, and it is expected to vary depending on the reactor as well as the fuel rod design. This value, while representing some relaxation from the assumption of instantaneous appearance, is nevertheless conservative. As noted in Reference 15, these estimates are valid for a double-ended rupture of the largest pipe, assume that the fuel rod is being operated at the maximum peaking factor permitted by the plant Technical Specifications and at the highest burnup levels anticipated, and assume that the emergency core cooling system (ECCS) is not operating. Use of more realistic assumptions for any of these parameters would increase estimated times to fuel rod failure by factors of two or more. Nevertheless, the use of conservative assumptions in estimating fuel rod failure times is considered appropriate since such failure times are likely to be used primarily in consideration of the necessary closure time for certain containment isolation valves. Since it is important that closure of such valves be ensured before the release of significant radioactivity to the environment, a conservative estimate of fuel failure time and consequent onset of fission product appearance is deemed appropriate. For plants with leak-before-break approval for their reactor coolant system piping, a longer duration before fuel clad failure is expected. However, other constraints may become the limiting factor on containment isolation valve closure time.

4.3 Release Phase Durations

The durations of the various release phases have been selected primarily by examination of the values

available for the group of severe accident scenarios considered in Section 3. The durations of the early in-vessel and ex-vessel release phases differs for BWRs versus PWRs and reflect the differing core heatup rates as well as the differing amounts of zirconium available to supply chemical energy after core-melt. While the selected durations of the release phases are realistic, some conservatisms should be noted. The duration of the early in-vessel release phase for BWRs and PWRs is short and does not represent a probabilistically weighted average or mean value for the accident sequences considered. This will introduce a given quantity of fission products into containment in a shorter time than might be expected for a typical sequence.

Similarly, the duration of the ex-vessel release phase, while considered realistic for the bulk of the fission products being released, is short for releases of tellurium and ruthenium since, as noted in Section 3.3, release of these nuclides occurs over a longer time.

The selected release duration times have been chosen primarily on the basis of simplicity, since an accurate determination of the duration of the release phases depends not only on the reactor type but also on the applicable accident sequence, which varies for each reactor design.

4.4 Composition and Magnitude of Releases

The composition of the fission products was initially based on the grouping developed with the STCP, but has been modified as discussed in Section 3.4.

The magnitudes of the fission products released into containment for the accident source term were selected in the draft version of this report to be the mean values, using NUREG-1150 methodology, for BWR and PWR low-pressure scenarios involving high estimates of zirconium oxidation. The uncertainty distributions for the in-vessel release and total release into containment are displayed graphically in Appendix A. Bounding estimates for the releases into containment taken from Reference 17, using the STCP methodology, are shown in Appendix B.

The release magnitudes for the low volatile fission products were reduced significantly in the final report. This reduction was based upon recent experimental research results (Ref. 25) since completion of NUREG-1150, as well as a re-examination of the uncertainty distribution, in response to comments on the draft report. Research on in-vessel phenomena includes the in-pile Severe Fuel Damage (SFD) experiments in the Power Burst Facility (PBF), further examination of the Three Mile Island (TMI) accident, and the SASCHA out-of-pile tests. Ex-vessel insights derive primarily from large scale tests performed as part of the internationally sponsored ACE Program. Reference 25 notes that, based on the SFD experiments as well as the TMI accident, in-vessel release fractions for cerium, for example, were about 10^{-4}, compared to the value of 10^{-2} cited in the draft report. Based on these results, the NRC staff concludes that the low volatile release fractions cited in draft NUREG-1465 are too high.

The uncertainty distributions were also examined to obtain additional insight. As can be seen from the uncertainty distributions in Appendix A, the range of release estimates for the volatile nuclides, such as the noble gases, iodine, cesium, and to some extent tellurium, spans about one order of magnitude. For this group of nuclides, use of the mean value is a reasonable estimate of the release fraction. In contrast, the range for the low volatile nuclides, such as barium, strontium, cerium and lanthanum, spans about 4 to 6 orders of magnitude. For the latter group of nuclides, the mean value can be misleading, since it may be well in excess of other measures of the distribution. This is illustrated in Table 4.1 which tabulates the mean, median, and 75th percentile values for several low volatile nuclides released during the early in-vessel phase.

Table 4.1 Measures of Low Volatile In-Vessel Release Fractions

Nuclide	Mean	Median	75th percentile
Sr	0.03	0.001	0.006
Ba	0.04	0.003	0.009
La	0.002	0.00003	0.0003
Ce	0.01	0.00006	0.0006

As can be seen from Table 4.1, the mean value for this group of nuclides is one to two orders of magnitude greater than the median value, and is about 5 times greater than the 75th percentile of the distribution. For this group of nuclides, the mean is controlled by the upper tail of the distribution, and the details of the whole distribution may be more indicative of the uncertainty than the "bottom line" results, such as a mean value. Because of this, the final version of this report has chosen not to use the mean value in estimating releases for the non-volatile nuclides. While the median value might be selected as an alternate, it fails to provide an appreciation of the range of values lying above it. Since this report is intended for regulatory applications, the intent is to avoid under-estimation of potential releases or offsite doses, without undue conservatism. Hence, for the final

report, the 75th percentile value has been selected for the low volatile nuclides on the basis that it bounds most of the range of values, without undue influence by the upper tail of the distribution.

Uncertainties, particularly in understanding and modeling core melt progression phenomena, can affect the duration of the early in-vessel release phase, including the timing of reactor pressure vessel failure. An increase in duration of the early in-vessel phase can lead to increased releases of volatile fission products during the early in-vessel phase and a concomitant reduction during the ex-vessel phase. An increase in duration of the early in-vessel phase, however, also provides additional time for fission product removal within containment by natural processes or fission product cleanup systems.

Upper bound estimates, tabulated in Appendix B, indicate that virtually all the iodine and cesium could enter the containment. Similarly, for tellurium, upper bound estimates indicate that as much as about two-thirds of the core inventory of tellurium could be released into containment. Hence, for this important group of radionuclides (iodine, cesium, and tellurium), the upper bound estimates of total release into containment are approximately 1.5 times the mean value estimates.

For the lower volatility radionuclides such as barium and strontium, upper bound estimates range from about 50 to 70% of the core inventory released into containment. Almost all of this is estimated to be released as a result of core-concrete interactions. In contrast, mean value estimates range from 15 to 25%. Hence, in this case, the upper bound estimates are about two to three times the mean values.

Finally, for the refractory nuclides such as lanthanum and cerium, the upper bound estimates indicate that about 5% of the inventory of these nuclides could appear within containment, whereas the mean value estimate indicates only about 1% released.

PRAs have indicated that, considering the magnitudes of the radioactive species estimated to be released to the environment for severe reactor accidents, the radionuclides having the greatest impact on risk are typically the volatile nuclides such as iodine and cesium, with tellurium to a somewhat lesser degree. The uncertainty distributions for this group of radionuclides is also the smallest, as shown in the graphical tabulations of Appendix A. Hence, our ability to predict the behavior and releases for this group of nuclides is significantly better than for other fission product groupings.

Mean value estimates selected for the in-containment accident source term provide reasonable estimates for the important nuclides consisting of iodine, cesium, and tellurium. These estimates show a relatively low degree of uncertainty and are unlikely to be exceeded by more than 50%. Uncertainty increases in estimating releases for the remaining nuclides.

4.5 Iodine Chemical Form

The chemical form of iodine entering containment was investigated in Reference 18. On the basis of this work, the NRC staff concludes that iodine entering containment from the reactor coolant system is composed of at least 95% cesium iodide (CsI), with no more than 5% I plus HI. Once within containment, highly soluble cesium iodide will readily dissolve in water pools and plate out on wet surfaces in ionic form. Radiation-induced conversion of the ionic form to elemental iodine will potentially be an important mechanism. If the pH is controlled to a level of 7 or greater, such conversion to elemental iodine will be minimal. If the pH is not controlled, however, a relatively large fraction (greater for PWRs than BWRs) of the iodine dissolved in containment pools in ionic form will be converted to elemental iodine.

5 IN-CONTAINMENT REMOVAL MECHANISMS

Since radioactive fission products within containment are in the form of gases and finely divided airborne particulates (aerosols), the principal mechanism by which fission products find their way from the reactor to the environment with an intact containment is via leakage from the containment atmosphere. The specific fission product inventory present in the containment atmosphere at any time depends on two factors: (1) the source, i.e., the rate at which fission products are being introduced into the containment atmosphere, and (2) the sink, the rate at which they are being removed. Aspects of the release and transport of fission products from the core into the containment atmosphere were presented in Section 3.

Mechanisms that remove fission products from the atmosphere with consequent mitigation of the in-containment source term fall into two classes: (1) engineered safety features (ESFs) and (2) natural processes. ESFs to remove or reduce fission products within the containment are presently required (Criterion 41 in Appendix A of 10 CFR Part 50) and include such systems as containment atmosphere sprays, BWR suppression pools, and filtration systems utilizing both particulate filters and charcoal adsorption beds for the removal of iodine, particularly in elemental form. Natural removal includes such processes as

aerosol deposition and the sorption of vapors on equipment and structural surfaces.

The draft version of this report contained a discussion of some of the more important fission product removal mechanisms, including some quantitative results. These numerical results were intended to be illustrative of the phenomena involved and were not intended to be applied rigorously, however. It was recognized that the data and illustrations used in the draft might not be applicable to all situations.

In recognition of this, the NRC staff undertook to examine, with contractor assistance, improved understanding of fission product removal mechanisms. At this time, this effort is still underway. Rather than provide numerical values that may be inapplicable, this report will provide references, where available, so that the reader may utilize improved methodologies to obtain results that apply to the situation at hand.

5.1 Containment Sprays

Containment sprays, covered in Standard Review Plan (SRP) Section 6.5.2 (Ref. 28), are used in many PWR designs to provide post-accident containment cooling as well as to remove released radioactive aerosols. Sprays are effective in reducing the airborne concentration of elemental and particulate iodines as well as other particulates, such as cesium, but are not effective in removing noble gases or organic forms of iodine. The reduction in airborne radioactivity within containment by a spray system as a function of time is expressed as an exponential reduction process, where the spray removal coefficient, lambda, is taken to be constant over a large part of the regime. Typical PWR containment spray systems are capable of rapidly reducing the concentration of airborne activity (by about 2 orders of magnitude within about 30 minutes, where both spray trains are operable). Once the bulk of the activity has been removed, however, the spray becomes significantly less effective in reducing the remaining fission products. This is usually accounted for by either employing a spray cut-off, wherein the spray removal becomes zero after some reduction has been achieved, or changing to a much smaller value of lambda to reflect the decreased removal effectiveness of the spray when airborne concentrations are low.

SRP Section 6.5.2 (Ref. 28) provides expressions for calculating spray lambdas, depending on plant parameters as well as the type of species removed. In addition, SRP 6.5.2 currently suggests that the containment sump solution be maintained at values at or above pH levels of 7, commencing with spray recirculation, to minimize revolatilization of iodine in the sump water. Current guidance states that containment spray systems be initiated automatically, because of the instantaneous appearance of the source term within containment, and that the spray duration not be less than 2 hours. In contrast, the revised source term information given in Section 3 suggests that spray system actuation might be somewhat delayed for radiological purposes, but that the spray system duration should be for a longer period of about 10 or more hours. Because sprays are effective in rapidly removing particulates from the containment atmosphere, intermittent operation over a prolonged period may also provide satisfactory mitigation.

The spray removal coefficient for particulates appears particularly important in view of the information presented in Section 3, which indicates that most fission products are expected to be in particulate form. The spray removal coefficient (λ) is derived from the following equation from Standard Review Plan Section 6.5.2

$$\lambda = \frac{3hFE}{2VD}$$

h = Fall height of spray drops
V = Containment building volume
F = Spray flow
E/D = the ratio of a dimensionless collection efficiency E to the average spray drop Diameter D. E/D is conservatively assumed to be equal to 10/meter for spray drops 1 mm in diameter changing to 1/meter when the aerosol mass has been depleted by a factor of 50.

Using values typical for PWRs, the formulation given in SRP 6.5.2 estimates particulate removal rates to be on the order of 5 per hour. Nourbakhsh (Ref. 29) examined the effectiveness of containment sprays, as evaluated in NUREG-1150 (Ref. 7), in decontaminating both in-vessel and ex-vessel releases. Powers and Burson (Ref. 30) have developed a more realistic, yet simplified, model with regard to evaluating the effectiveness of aerosol removal by containment sprays

5.2 BWR Suppression Pools

BWRs use pressure suppression pools to condense steam resulting from a loss-of-coolant accident. Prior to the release to the reactor building, these pools also scrub radioactive fission products that accompany the steam. Regulatory Guide 1.3 (Ref. 2) suggests not allowing credit for fission product scrubbing by BWR suppression pools, but SRP Section 6.5.5 (Ref. 31) was revised to suggest allowing such credit. The pool water will retain soluble, gaseous, and solid fission products such as iodines and cesium but provide no attenuation of the noble gases released from the core. The Reactor Safety Study (WASH-1400, Ref. 5) assumed a decontamination factor (DF) of 100 for subcooled

suppression pools and 1.0 for steam saturated pools. Since 1975 when WASH-1400 was published, several detailed models have been developed for the removal of radioactive aerosols during steam flow through suppression pools.

Calculations for a BWR with a Mark I containment (Ref. 27) used in NUREG-1150 (Ref. 7) indicate that DFs ranged from 1.2 to about 4000 with a median value of about 80. The suppression pool has been shown to be effective in scrubbing some of the most important radionuclides such as iodine, cesium, and tellurium, as these are released in the early in-vessel phase. The NRC staff is also presently reviewing fission product scrubbing by suppression pools to develop simplified models.

If not bypassed, the suppression pool will also be effective in scrubbing ex-vessel releases. Suppression pool bypass is an important aspect that places an upper limit on the overall performance of the suppression pool in scrubbing fission products. For example, if as little as 1% of the fission products bypass the suppression pool, the effective DF, taking bypass into account, will be less than 100, regardless of the pool's ability to scrub fission products.

Although decontamination factors for the suppression pool are significant, the potential for iodine re-evolution can be important. Re-evolution of iodine was judged to be important in accident sequences where the containment had failed and the suppression pool was boiling. There is presently no requirement for pH control in BWR suppression pools. Hence, it is possible that suppression pools would scrub substantial amounts of iodine in the early phases of an accident, only to re-evolve it later as elemental iodine. It may well be that additional materials likely to be in the suppression pool as a result of a severe accident, such as cesium borate or cesium hydroxide and core-concrete decomposition products, would counteract any reduction in pH from radiolysis and would ensure that the pH level was sufficiently high to preclude re-evolution of elemental iodine. Therefore, if credit is to be given for long-term retention of iodine in the suppression pool, maintenance of the pH at or above a level of 7 must be demonstrated. It is important to note, however, that this is not a matter of concern for present plants since all BWRs employ safety-related filtration systems (see Section 5.3) designed to cope with large quantities of elemental iodine. Hence, even if the suppression pool were to re-evolve significant amounts of elemental iodine, it would be retained by the existing downstream filtration system.

5.3 Filtration Systems

ESF filtration systems are discussed in Regulatory Guide 1.52 (Ref. 32) and are used to reduce the radioactive aerosols and iodine released during postulated accident conditions.

A typical ESF filtration system consists of redundant trains that each have demisters to remove steam and water droplets from the air entering the filter bank, heaters to reduce the relative humidity of the air, high efficiency particulate air (HEPA) filters to remove particulates, charcoal adsorbers to remove iodine in elemental and organic form, followed finally by additional HEPA filters to remove any charcoal fines released.

Charcoal adsorber beds can be designed, as indicated in Regulatory Guide 1.52, to remove from 90 to 99% of the elemental iodine and from 30 to 99% of the organic iodide, depending upon the specific filter train design.

Revised insights on accident source terms, given in Section 3, may have several implications for ESF filtration systems. Present ESF filtration systems are not sized to handle the mass loadings of non-radioactive aerosols that might be released as a result of the ex-vessel release phase, which could produce releases of significant quantities of nonradioactive as well as radioactive aerosols. However, if ESF filtration systems are employed in conjunction with BWR suppression pools or if significant quantities of water are overlaying molten core debris (see Section 5.4), large quantities of nonradioactive (as well as radioactive) aerosols will be scrubbed and retained by these water sources, thereby reducing the aerosol mass loads upon the filter system.

A second implication of revised source term insights for ESF filtration systems is the impact of revised understanding of the chemical form of iodine within containment. Present ESF filtration systems presume that the chemical form of iodine is primarily elemental iodine, and these systems include charcoal adsorber beds to trap and retain elemental iodine. Assuming that pH control is maintained within the containment, a key question is whether charcoal beds are necessary. Two questions appear to have a bearing on this issue and must be addressed, even assuming pH control. These are (1) to what degree will CsI retained on particulate filters decompose to evolve elemental iodine? and (2) what effect would hydrogen burns have on the chemical form of the iodine within containment? Based on preliminary information, CsI retained on particulate filters as an aerosol appears to be chemically stable provided that it is not exposed to moisture. Exposure to moisture, however, would lead to CsI decomposition and production of iodine in ionic form (I^-), which in turn would lead to re-evolution of elemental iodine. Although ESF filtration systems are equipped with demisters and heaters to remove significant moisture before it reaches the charcoal adsorber bed, an

additional concern is that the demisters themselves may trap some CsI aerosol.

In conclusion, present ESF filtration systems, while optimized to remove iodine, particularly in elemental form, have HEPA filters that are effective in the removal of particulates as well. Although such filtration systems are not designed to handle the large mass loadings expected as a result of ex-vessel releases, when they are used in conjunction with large water sources such as BWR suppression pools or significant water depths overlaying core debris, the water sources will reduce the aerosol mass loading on the filter system significantly, making such filter systems effective in mitigation of a large spectrum of accident sequences.

5.4 Water Overlying Core Debris

Experimental measurements (Ref. 33) have shown that significant depths of water overlying any molten core debris after reactor pressure vessel failure will scrub and retain particulate fission products. The question of coolability of the molten debris as a result of water overlying it is still under investigation. A major factor that may affect the degree of scrubbing is whether the water layer in contact with the molten debris is boiling or not.

Results from Ref. 33 indicate that both subcooled as well as boiling water layers having a depth of about 3 meters had measured DFs of about 10. A recent study (Ref. 34) performed for the NRC has provided a simplified model to determine the degree of aerosol scrubbing by a water pool overlying core debris interacting with concrete.

5.5 Aerosol Deposition

Since the principal pathway for transport of fission products is via airborne particulates, i.e., aerosols, this subject is discussed in some detail. Aerosols are usually thought of as solid particulates, but in general, the term also includes finely divided liquid droplets such as water, i.e., fog. The two major sources of aerosols are condensation and entrainment. Condensation aerosols form when a vapor originating from some high-temperature source moves into a cooler region where the vapor falls below its saturation temperature and nucleation begins. Entrainment aerosols form when gas bubbles break through a liquid surface and drag droplets of the liquid phase into the wake of the bubble as it leaves the surface. In general, condensation particles are smaller in size (submicron to a few microns), while entrainment particles are usually larger (1.0–100 microns). Once airborne, both types of aerosols behave in a similar manner with respect to both natural and engineered removal processes.

There are four natural processes that remove aerosols from the containment atmosphere over a period of time: (1) gravitational settling, (2) diffusiophoresis, (3) thermophoresis, and (4) particle diffusion. (Particle diffusion is less important than the first three processes and will not be discussed further.) All particles fall naturally under the force of gravity and collect on any available surface that terminates the fall, e.g., the floor or upper surfaces of equipment. Both diffusiophoresis and thermophoresis cause the deposition of aerosol particles on all surfaces regardless of their orientation, i.e., walls and ceiling as well as the floor. Diffusiophoresis is the process by which water vapor in the atmosphere 'drags' aerosol particles with it as it migrates (diffuses) toward a relatively cold surface on which condensation is taking place. Thermophoresis also causes aerosol particles to move toward and deposit on colder surfaces but not as a result of mass motion. Rather, the decreasing average velocity of the surrounding gas molecules tends to drive the particle down the temperature gradient until it traverses the interface layer and comes into contact with the surface where it sticks.

Aerosol agglomeration is another natural phenomenon that has an influence on the rates at which the removal processes described above will proceed. Agglomeration results from the random inelastic collisions of particles with each other. The process brings about a gradual increase in average particle size resulting in more rapid gravitational settling. Three phenomena contribute to particle growth by agglomeration: (1) Brownian motion, (2) gravitational fall, and (3) turbulence. Brownian agglomeration is caused by particle collisions resulting from random 'buffeting' by high-energy gas molecules. Gravitational agglomeration results from the fact that some particles fall faster than others and therefore tend to collide with and stick to other slower falling particles on their way down. Finally, rapid variations in gas velocity and flow direction in the atmosphere, i.e., turbulence, tend to increase the rate at which particle collisions occur and thus increase the average particle size. It is to be expected that, as agglomeration advances, the size of the particle will increase, and its shape can be expected to change as well. These latter factors have a strong influence on the removal processes.

The agglomeration and aerosol removal processes all depend critically upon the thermodynamic state and thermal-hydraulic conditions of the containment atmosphere. For example, the condensation onto and evaporation of water from the aerosol particles themselves have strong effects on all of the agglomeration and removal processes. Water condensed on aerosol particles increases their mass and makes them more spherical; both of these effects tend to increase the rate of gravitational settling. Some

aerosols, such as CsI and CsOH, are hygroscopic and absorb water vapor even when the containment atmosphere is below saturation. As with condensation, hygroscopicity also increases the rate of deposition.

Because of its importance to fields such as weather and atmosphere pollution, the behavior of aerosols has been under study for many decades. A number of computer codes have been developed to specifically consider aerosol behavior as it relates to nuclear accident conditions. The most complete mechanistic treatment of aerosol behavior in the reactor containment is found in CONTAIN, a computer code developed at Sandia National Laboratories under NRC sponsorship for the analysis of containment behavior under severe accident conditions. The aerosol models in the NAUA code are very similar to those used in CONTAIN; NAUA was developed at the Kernforschungszentrum, Karlsrhue, F.R.G., and was used for aerosol treatment in the NRC STCP. There are a number of other well-known aerosol behavior computer codes, but these two are the most widely used and accepted throughout the international nuclear safety community.

The rate at which gravitational settling occurs depends upon the degree of agglomeration at any particular time (i.e., the average particle size) as well as the total particle density m (mass per unit volume). Thus, as in most cases where the decrement of a variable is proportional to the variable itself, one can expect an exponential behavior. The gravitational settling process is quite complex and depends upon a large number of physical quantities, e.g., collision shape factor, particle settling shape factor, gas viscosity, effective settling height, density correction factor, normalized Brownian collision coefficient, gravitational acceleration, and particle material density. The only variable in this list that is independent of the plant, the accident scenario, and the atmospheric thermal-hydraulic conditions is the constant of gravitation. It follows that no single DF can be ascribed to cover the entire range of plant designs, accident scenarios, and source materials. An effort is under way to establish a set of simplified algorithms that can be used to provide a set of specific ranges of atmosphere conditions. This effort is still underway at this time.

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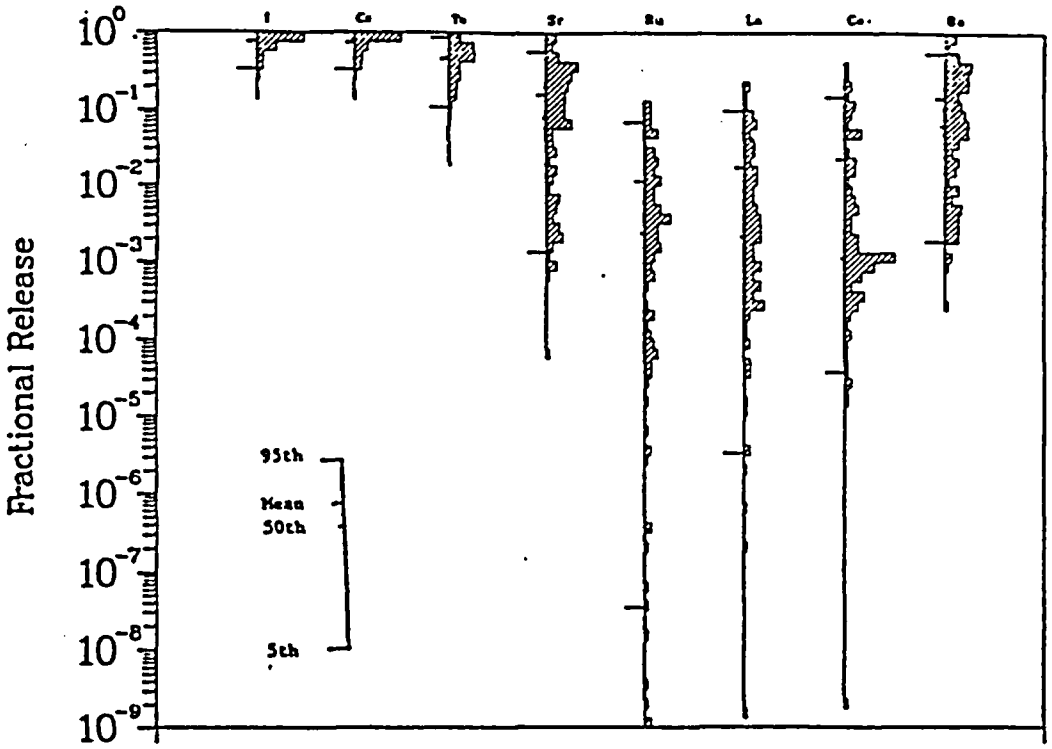
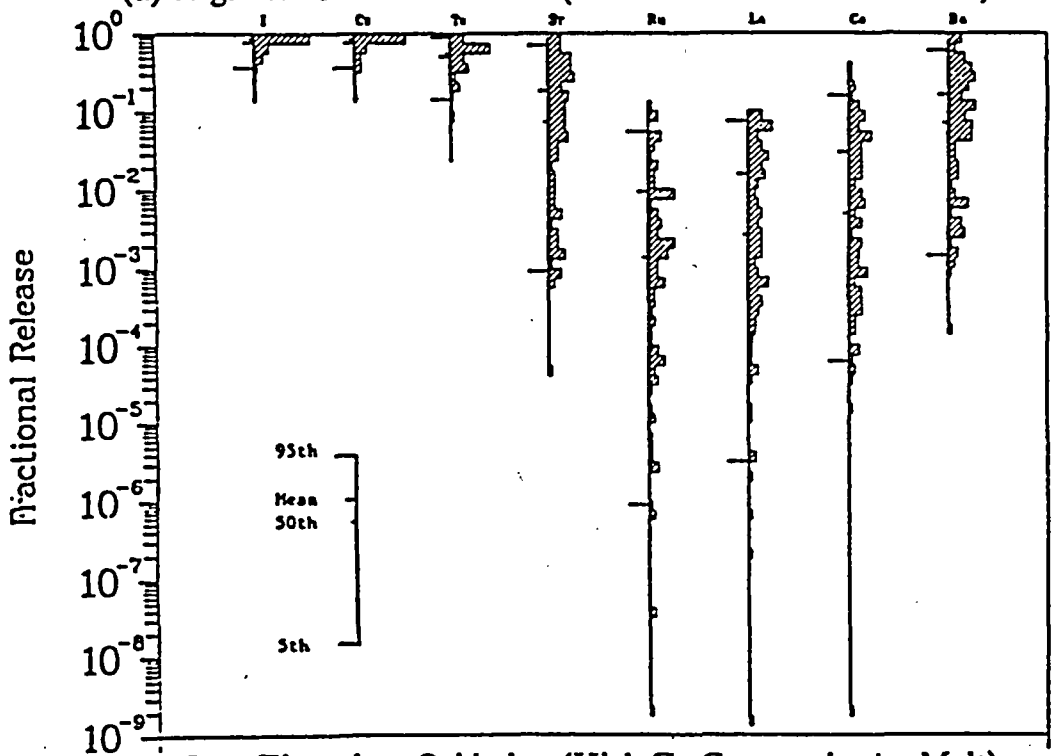
APPENDIX A
UNCERTAINTY DISTRIBUTIONS

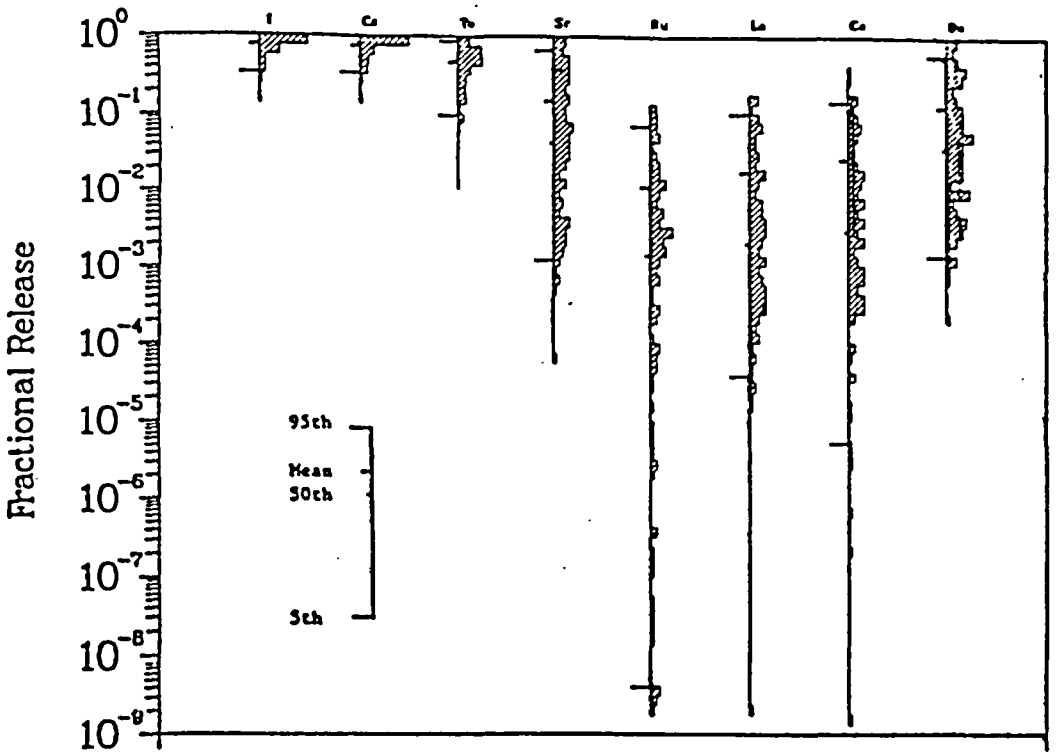


(a) High Zirconium Oxidation (Low Zr Content in the Melt)



(b) Low Zirconium Oxidation (High Zr Content in the Melt)

Uncertainty Distributions for Total Releases Into Containment PWR, Low RCS Pressure, Limestone Concrete, Dry Cavity, Two Openings After VB, FPART = 1.



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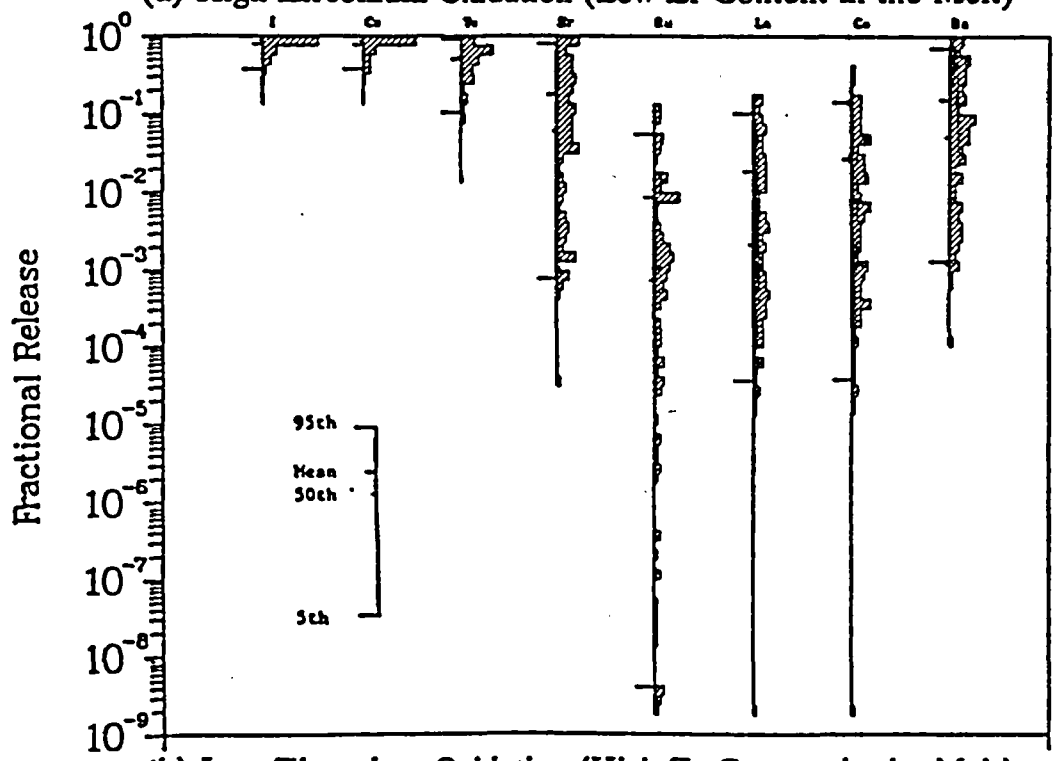
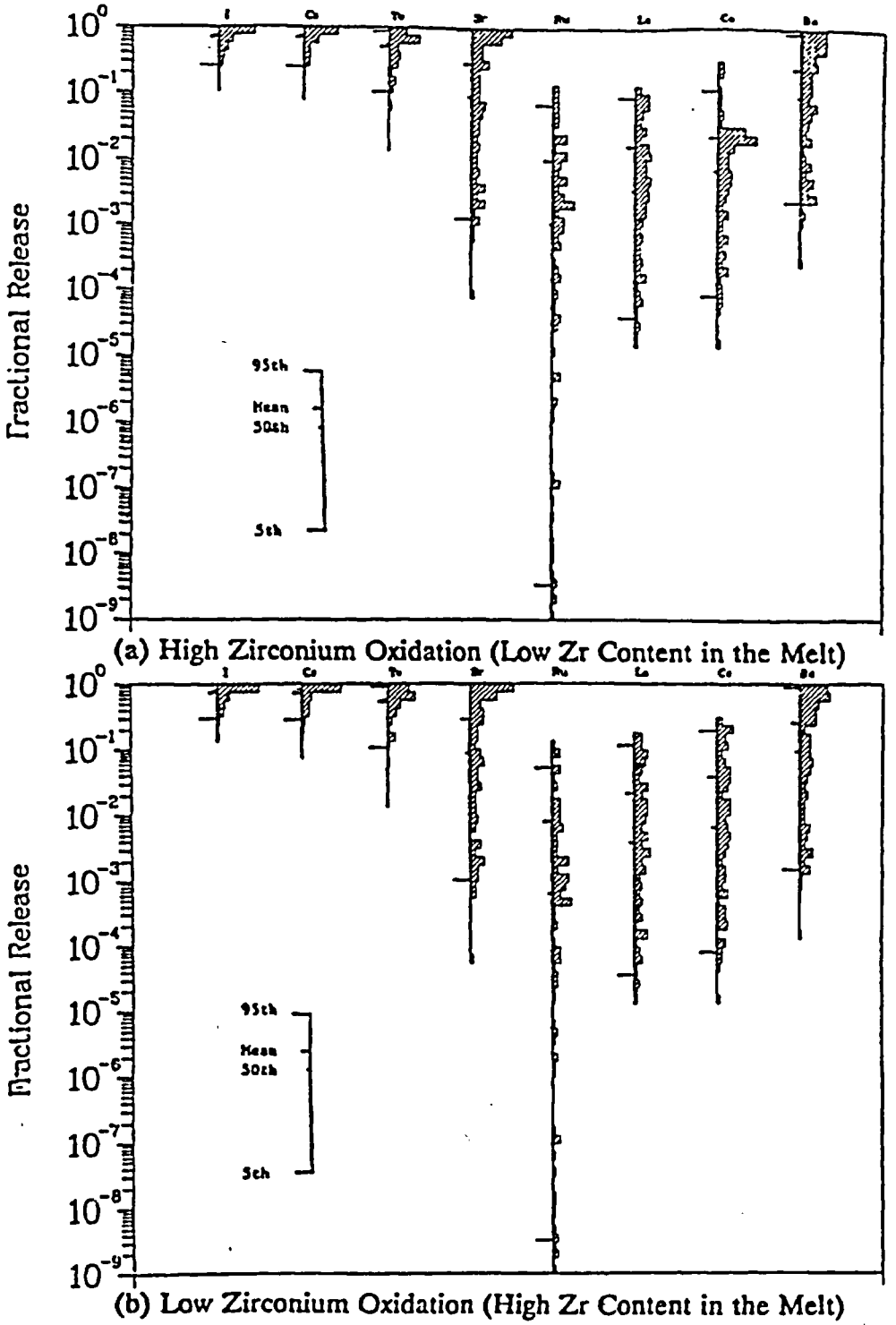


(b) Low Zirconium Oxidation (High Zr Content in the Melt)

Uncertainty Distributions for Total Releases Into Containment PWR, Low RCS Pressure, Basaltic Concrete, Dry Cavity, Two Openings After VB, FPART = 1.



Uncertainty Distributions for Total Releases Into Containment BWR, Low Pressure Fast Station Blackout, Limestone Concrete, Dry Pedestal, Low Drywell Temperature, FPART = 1.

APPENDIX B

STCP BOUNDING VALUE RELEASES

Updated Bounding Value of Radionuclide Releases into the Containment Under Severe Accident Conditions for PWRs

	ST_{INV}[a]		ST_{VB}	ST_{EXV}[b]		ST_{REV}	
	High RCS Pressure	Low RCS Pressure	High RCS Pressure	Limestone Concrete	Basaltic Concrete	High RCS Pressure	Low RCS Pressure
NG	1.0	1.0	0.	0.	0.	0.	0.
I	0.30	0.75	0.10	0.15	0.15	0.05	0.02
Cs	0.30	0.75	0.10	0.15	0.15	0.02	0.02
Te	0.20	0.50	0.05	0.40	0.30	0.02	0.01
Sr-Ba	0.003	0.01	0.01	0.40	0.15	–	–
Ru	0.003	0.01	0.05	0.005	0.005	–	–
La-Ce	5×10^{-5}	1.5×10^{-4}	0.01	0.05	0.05	–	–
Release Duration	40 minutes			2 hours[c]		10 hours	

[a] All entries are fractions of initial core inventory.

[b] Assuming 100% of the core participate in CCI.

[c] Except for Te and Ru where the duration is extended to five hours.

Updated Bounding Value of Radionuclide Releases into the Containment Under Severe Accident Conditions for BWRs

	ST_{INV}[a]		ST_{VB}	ST_{EXV}[c]		ST_{REV}	
	High RCS Pressure	Low RCS Pressure[b]	High RCS Pressure	Limestone Concrete	Basaltic Concrete	High RCS Pressure	Low RCS Pressure[b]
NG	1.	1.	0.	0.	0.	0.	0.
I	0.50	0.75	0.10	0.15	0.15	0.10	0.02
Cs	0.50	0.75	0.10	0.15	0.15	0.05	0.01
Te	0.10	0.15	0.05	0.50	0.30	0.02	0.02
Sr-Ba	0.003	0.01	0.01	0.70	0.30	–	–
Ru	0.003	0.01	0.05	0.005	0.005	–	–
La-Ce	5×10^{-5}	1.5×10^{-4}	0.01	0.10	0.10	–	–
Release Duration	1.5 hours			3 hours[d]		10 hours	

[a] All entries are fractions of initial core inventory.

[b] High pressure ATWS are also considered in this category.

[c] Assuming 100% of the core participate in CCI.

[d] Except for Te and Ru where the duration is extended to six hours.

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11. ABSTRACT *(200 words or less)*

In 1962 the U.S. Atomic Energy Commission published TID-14844, "Calculation of Distance Factors for Power and Test Reactors" which specified a release of fission products from the core to the reactor containment for a postulated accident involving "substantial meltdown of the core". This "source term", the basis for the NRC's Regulatory Guides 1.3 and 1.4, has been used to determine compliance with the NRC's reactor site criteria, 10 CFR Part 100, and to evaluate other important plant performance requirements. During the past 30 years substantial additional information on fission product releases has been developed based on significant severe accident research. This document utilizes this research by providing more realistic estimates of the "source term" release into containment, in terms of timing, nuclide types, quantities and chemical form, given a severe core-melt accident. This revised "source term" is to be applied to the design of future light water reactors (LWRs). Current LWR licensees may voluntarily propose applications based upon it.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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Core Meltdown
Design Basis Accident
TID-14844 Replacement
Core Fission Product Releases

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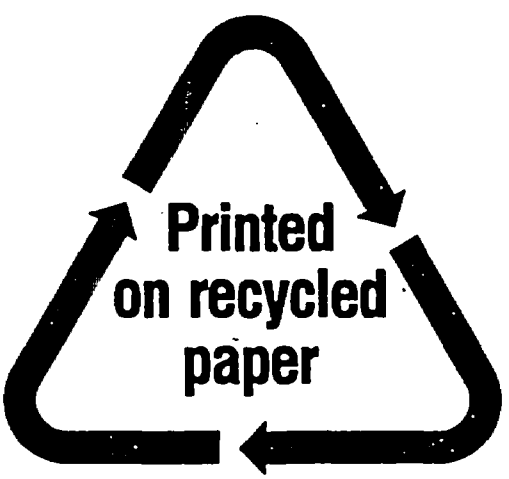

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ATTACHMENT 9

Excerpt from NRC Office of Nuclear Regulatory Research, *Draft for Comment, Reactor Risk Reference Document*, NUREG-1150, Vol. 1 (Feb. 1987)

NUREG-1150
Vol. 1

Reactor Risk Reference Document

Main Report

Draft for Comment

Manuscript Completed: January 1987
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Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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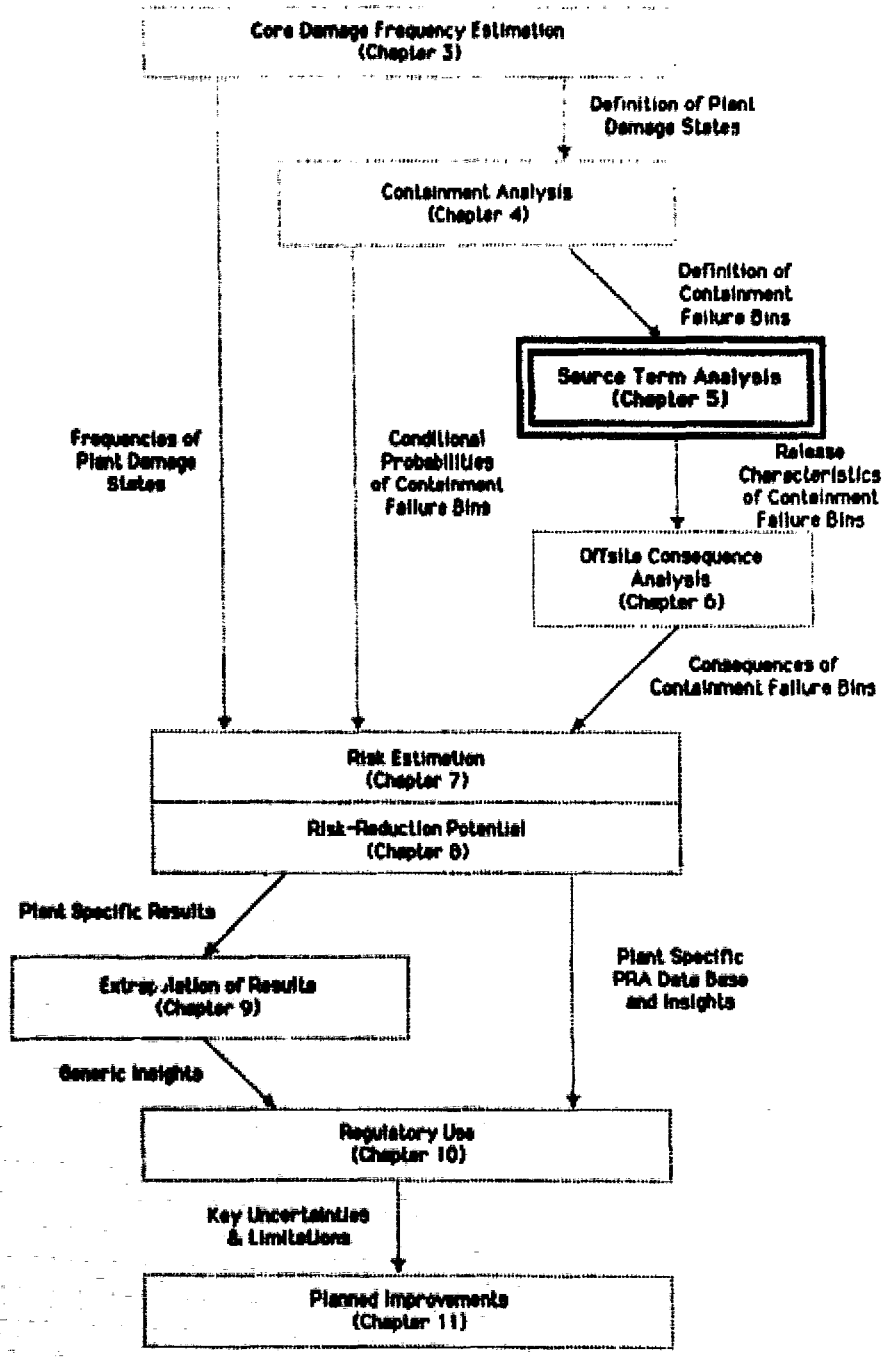
5. SOURCE TERM ANALYSIS



5.1 Introduction

The amount and timing of the release of radioactive material to the environment in an accident is referred to as a source term. Source terms are the input to ex-plant consequence analysis codes such as CRAC2 (Ref. 5.1) or MACCS (Ref. 5.2). Because many of the end states of the containment event tree would have very

similar source terms, they are grouped into containment failure bins. A set of radionuclide release fractions must therefore be determined for each bin, as illustrated in the above figure. The characteristics of the containment failure bins are determined in the containment event tree task and the release fractions are then determined in the source term task. In practice, the process is iterative.

Source terms are typically characterized by the fractions of the core inventory of radionuclides that are released to the environment, as well as the time dependence of the release, the size distribution of the aerosols released, the elevation of the release, the time of containment failure, the warning time, and the energy released with the radioactive material, all of which are required for input to the consequence codes.

Shortly after the accident at Three Mile Island, the NRC initiated a program to review the adequacy of the methods available for predicting the magnitude of source terms for severe reactor accidents. After considerable effort and extensive peer review, the NRC published a report entitled "Reassessment of the Technical Bases for Estimating Source Terms," NUREG-0956 (Ref. 5.3), which describes a consistent and integrated approach to estimating source terms. The report recommends that a set of coupled computer codes, the Source Term Code Package (Ref. 5.4), be used as the state-of-the-art methodology for source term analysis at this time. These are the methods that have been used as the principal basis for source term estimates for this study of accident risk. Since a separate source term result is required for each different combination of the variables in the statistical sampling analysis and containment failure bin, it is not practical to perform a Source Term Code Package calculation for each combination of the variables of interest. For this reason, simplified methods of analysis with adjustable parameters that were determined from Source Term Code Package results were developed. In addition, the simplified source term methods include a parametric representation of a number of source term issues that are not treated mechanistically in the Source Term Code Package but are varied in the statistical sampling analysis. Thus, the simplified source term methods not only were used to extend results obtained with the Source Term Code Package to different plant conditions, but they also played a key role in the performance of uncertainty analyses (Ref. 5.5).

5.2 Results for Surry Power Station Unit 1

5.2.1 Ranges of Source Term Results

Twelve source term issues were considered in the statistical sampling analysis for the Surry plant. The uncertainty assessments that had been made in the Quantitative Uncertainty Estimate of the Source Term (QUEST) program (Ref. 5.6) were frequently used to assist in determining alternative issue levels.

In-Vessel Release from Fuel

The uncertainties in the release from fuel in-vessel are not only the result of uncertainties associated with the migration of radioactive materials within the fuel and their release from the surface of the fuel but also with the details of melt progression. Four levels of release were considered: low, base (which was based on Source Term Code Package results), high, and very high. Each

level was represented by a set of release fractions for each of the seven elemental groups of radionuclides.

Amount of Cesium Iodide Decomposition

Although chemical equilibrium analyses (Ref. 5.7) indicate that cesium iodide (CsI) would be the predominant form of iodine in simple systems of cesium, iodine, steam, and hydrogen under in-vessel core melt conditions, a number of processes could decompose CsI to form more volatile species. These species would be more likely to escape the reactor coolant system. The range of decomposition of CsI assumed was from 0 to 100 percent, and the weights were fairly uniform indicating a high degree of uncertainty among the review-group members.

Retention in Reactor Coolant System

Four levels of retention were considered: high, base (based on the Source Term Code Package), low, and very low. For each set, different retention factors were defined for iodine/cesium, tellurium, and the less volatile radionuclides. Separate sets were developed for high-pressure and for low-pressure sequences.

Decontamination Factor for V Sequence

For those scenarios in which the point of release is submerged, it is necessary to estimate the pool decontamination factor. Since the depth of submergence, failure size, and orientation of leak are not well defined, the range of possible decontamination factors, from 2 to 100, is quite broad.

Magnitude of Core-Concrete Interaction Release

In the Source Term Code Package, the magnitude of the ex-vessel release is calculated by the CORCON/VANESA modules. Typically, these modules predict higher releases of radionuclides than the methods used by industry. As a result, the area has been the subject of considerable technical dispute. The mode of vessel failure, degree of dispersal of fuel debris, and time-dependence of release of fuel to the cavity are uncertain parameters that also influence the size of the release. Four levels of release were considered in the analysis with release terms that varied by as much as 100.

Decontamination Factors for Containment Sprays

The effectiveness of sprays in the removal of aerosols is very sensitive to the size distributions of the aerosols and the spray droplets. Three cases were considered: (1) effect on release from the reactor coolant system in high-pressure sequences in which containment failure follows vessel meltthrough, (2) effect on release from reactor coolant system in other sequences, and (3) effect on core-concrete interaction release. The range of decontamination factors for release from the reactor coolant system is from 5 to 100 and for the core-concrete release from 10 to 1,000.

Aerosol Agglomeration Uncertainties

Although aerosol transport, agglomeration, and deposition processes are well theoretically characterized and experimentally demonstrated, some aspects of

the aerosols must be specified that are difficult to predict such as aerosol shape factors. The range of aerosol characteristics used in the analyses were primarily based on studies undertaken in the QUEST uncertainty analysis program (Ref. 5.6).

Late Iodine Release from Containment

After containment failure, iodine that has been dissolved in water on the containment floor will evolve from the pool, and volatile organic iodides may be formed on containment surfaces leading to an extended period of release from the containment. The range of release considered in the statistical analysis was from zero to 10 percent of the inventory of iodine in the containment.

Late Revolatilization from Reactor Coolant System

Volatile radionuclides deposited on reactor coolant system surfaces early in an accident may be revolatilized later as the surfaces are heated. The Source Term Code Package models the process of revolatilization to the time of vessel meltthrough. After meltthrough, however, the uncertainties in the processes controlling revolatilization increase. The amount of revolatilization is influenced by complex natural convection flow patterns in the reactor coolant system, the extent of degradation of the reactor coolant system insulation, and the chemistry of the interaction of the radionuclides with the surface and other contaminants on the surface. The range of release considered included up to 70 percent of the amount of iodine and the cesium deposited on the surfaces.

Releases Associated With High-Pressure Ejection and Direct Containment Heating

When the reactor coolant system is at elevated pressure at the time of vessel meltthrough, significant aerosol formation is expected with the expulsion of molten core material, even if the dispersed core debris does not undergo oxidation resulting in significant pressurization of the containment. Under such conditions, radioactive releases can be enhanced. In this study, this release is divided into a high-pressure ejection and a direct containment heating release. Four levels of release were developed for both mechanisms. The enhanced release was assumed to occur for the fraction of the core ejected in the case of the high-pressure ejection component. The additional release associated with the direct heating component was assumed to only affect the fraction of this material that participated in direct containment heating.

As described in Section 4.2.2, 19 containment failure bins were defined for the Surry plant to represent the principal end states of the containment event tree. In the statistical sampling analysis that was performed to develop the uncertainty range for the plant risk, there is potentially a separate source term analysis required for each combination of containment failure bin and statistical sample member, since, in each member, variations are made in assumptions that affect the course of the accident and the mechanisms that affect the release of radionuclides. Figure 5.1 shows the range of results obtained for some selected bins. These ranges are compared with results obtained with the suite of codes in the Source Term Code Package (the Surry results were obtained prior to the completion of the code package) (Ref. 5.8). The Source Term Code Package results are not to be considered best estimates because they do not account for a number of important source term issues, which can lead to either higher or lower source terms. A single value, rather than a range, is presented

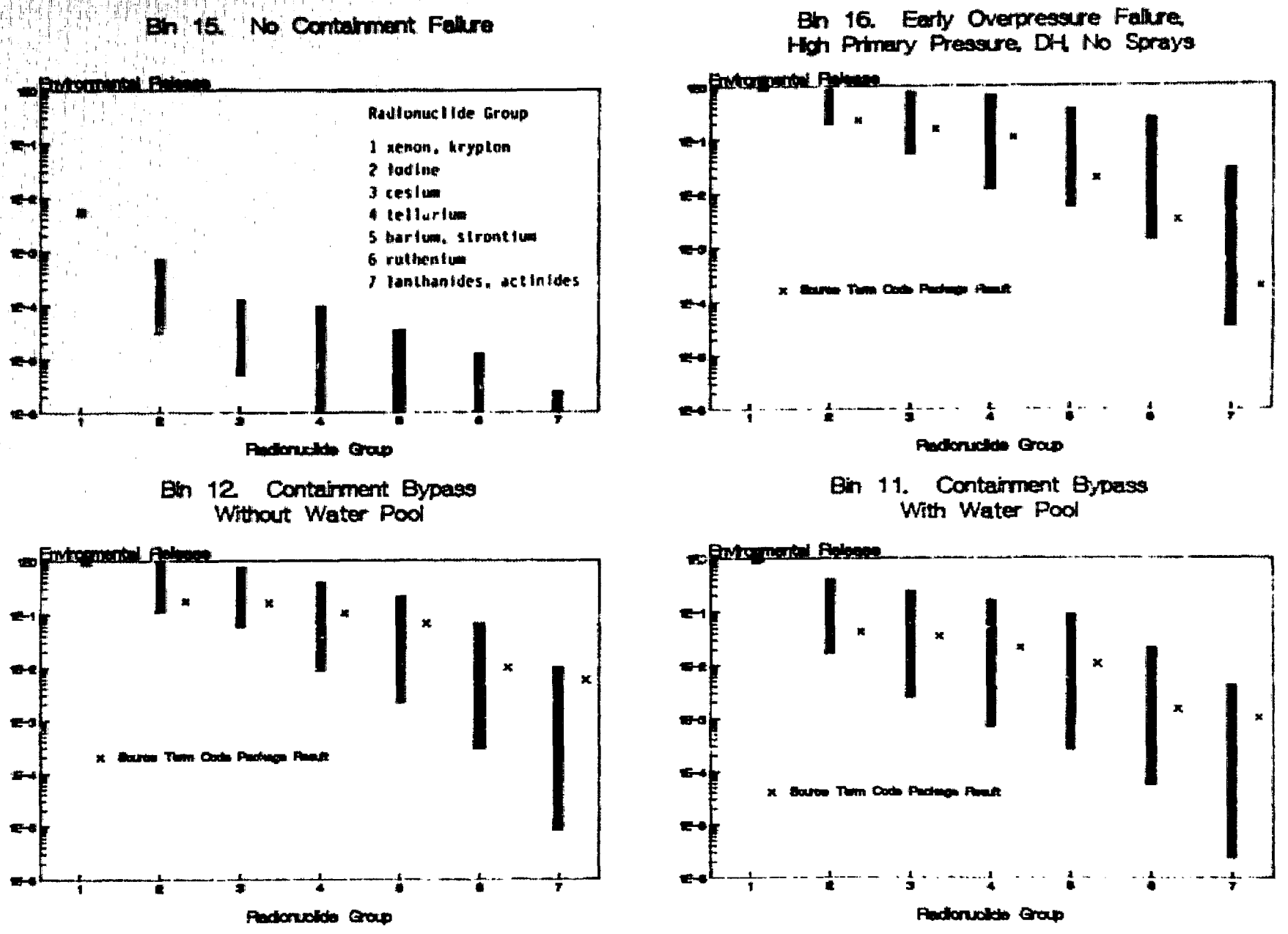


Figure 5.1 Ranges of release fractions for selected bins at Surry

for the noble gases because, in sequences in which the containment fails, essentially all the noble gases will be released to the environment.

For the volatile radionuclides (the iodine, cesium, and tellurium groups), the range of uncertainty varies from one to two orders of magnitude. For the nonvolatile radionuclides (the barium, ruthenium, and lanthanum groups), the range of uncertainty varies by two to three orders of magnitude. Bin 15, in which the containment leaks at the design value but does not fail in the accident, has the lowest source terms of the containment failure bins. These results are essentially the same as those for Bin 14 in which there is meltthrough of the basemat but no aboveground failure of the containment. The iodine group (Elemental Group 2) has somewhat higher releases than the cesium group (Elemental Group 3) because of the formation of volatile chemical forms of iodine over an extended period of time. Indeed for all the bins, the iodine release band is higher than the cesium band because of mechanisms that can lead to the formation of volatile forms of iodine and the reevolution of iodine from pools of water.

Bin 16 involves early failure by direct heating pressurization of the containment. If one assumes that direct heating occurs sufficient to threaten containment integrity and accounts for the frequencies of the plant damage states, Bin 16 is the most likely early failure mode of the containment. The release fractions are typically four orders of magnitude higher than for the no-failure case. The comparable bin without direct heating is Bin 1, which is illustrated in Figure 5.2. The ranges of release fractions for the more volatile groups of radioactive materials are the same for the two bins. The range of release fractions for the barium/strontium group (Elemental Group 5) is shifted upward by approximately a factor of two for the direct heating case. For both the ruthenium and the lanthanum groups (Elemental Groups 6 and 7), the range of release fractions is shifted upward by a factor of three. A single Source Term Code Package run was performed to represent both the Bin 1 and Bin 16 release terms. This calculation is actually more representative of Bin 1 than Bin 16, since the Source Term Code Package does not model the enhanced release of radionuclides associated with direct heating.

Located in Bins 11 and 12 are the containment bypass source terms in which the release point in the safeguards building is either submerged beneath a pool of water or is above water. The effect of the decontamination in the water pool is not only to shift the band toward smaller releases but also to increase the spread of uncertainty since the effectiveness of scrubbing could vary over a broad range. Although pool decontamination is clearly beneficial in reducing the source term for this type of accident scenario, the uncertainty bands for the source terms are so wide that the overall perspective of the source term is not dramatically changed. For example, even with scrubbing by a water pool in the safeguards building, the iodine release fraction could still exceed 20 percent of the core inventory within the uncertainties associated with phenomenological issues.

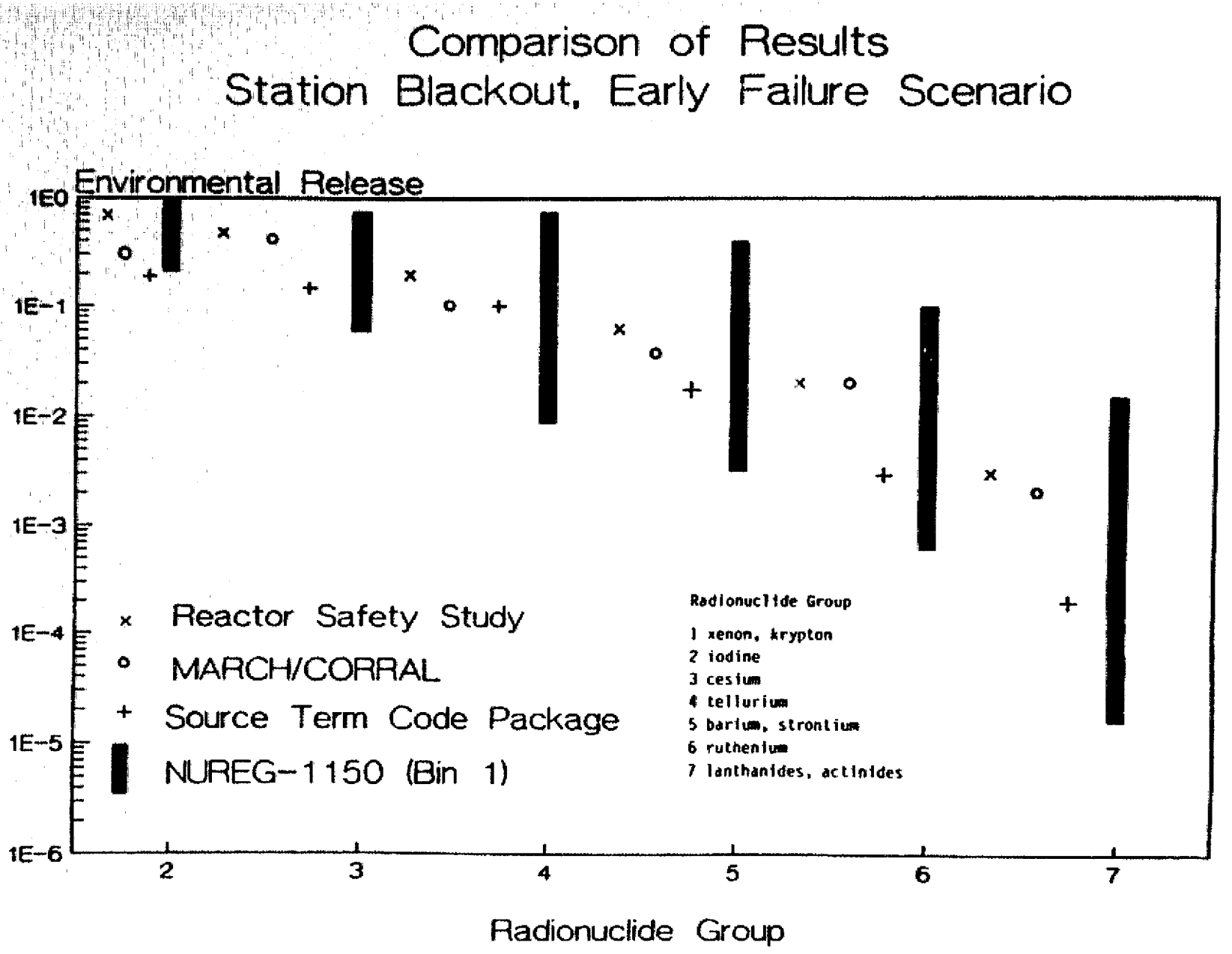


Figure 5.2 Comparison of results for station blackout scenarios at Surry

5.2.2 Comparison With Other Studies

Although probabilistic risk assessments (PRAs) use similar terminology to describe accident sequences, scenarios with the same identifier in two different studies frequently involve substantially different definitions and assumptions. What appear to be minor differences in assumptions can have a major impact on the calculated source term. For example, a few psi difference in assumed failure pressure could be the difference between early containment failure and late containment failure in an accident scenario in the Surry plant, with orders of magnitude variations in source terms. Furthermore, differences in the predicted behavior in one part of the analysis can propagate through the remainder of the analysis. In addition, the large uncertainties in the source term methods must be recognized when comparing point-estimate results. For example, a difference of a factor of two between source terms is certainly minor when interpreted within the context of two orders of magnitude uncertainty.

The treatment of source terms in PRAs has passed through three major phases as the capability to model severe accident processes has improved. The first phase was based on Reactor Safety Study (Ref. 5.9) source term results. The Reactor Safety Study analyzed two plants, Surry and Peach Bottom. The available data base was very sparse and the methods of analysis were crude. The next phase of source term analysis followed the writing of the MARCH code (Ref. 5.10). The combination of the thermal-hydraulic analysis capability of MARCH and the containment transport analysis capability of CORRAL (Ref. 5.11) permitted accident sequences to be analyzed from beginning to end. The radionuclide modeling assumptions were essentially identical to those used in the Reactor Safety Study. The core meltdown modeling assumptions were quite simple, relying heavily on conservation laws and intuition about the expected behavior of core meltdown progression. The MARCH/CORRAL methodology was used in the Reactor Safety Study Methodology Applications Program in the analysis of the Oconee, Calvert Cliffs, Sequoyah, and Grand Gulf plants (Refs. 5.12 and 5.13). A number of sequences were also reanalyzed for the Surry and Peach Bottom plants using these methods (Ref. 5.14).

Subsequent to the Three Mile Island accident, the NRC undertook a comprehensive research program to develop and validate methods for the analysis of severe accident phenomena (Ref. 5.15). This program has been augmented by contributions from the Electric Power Research Institute and IDCOR programs in the United States and cooperative programs with a number of foreign countries. One of the products of this research program has been the Source Term Code Package used in this study. The technical basis for the Source Term Code Package is described in NUREG-0956 (Ref. 5.3).

At the same time as the NRC has been developing a complement of computer codes for analyzing severe accidents, the nuclear industry has also been developing analysis capabilities. The analog to the Source Term Code Package is the IDCOR Modular Accident Analysis Program (MAAP) (Ref. 5.16). Although there are a number of differences in modeling assumptions in the two code packages, both packages were developed with similar objectives. Each attempts to perform a consistent, realistic analysis of severe accident processes. However, to date the MAAP program has not yet been subjected to the same level of external peer review as the Source Term Code Package.

Figure 5.2 compares source terms that have been obtained in different studies for the station blackout sequence in the Surry plant. The mode of containment failure is by steam spike and/or hydrogen deflagration. Results are compared for the Reactor Safety Study, the Reactor Safety Study Methodology Applications Program using the MARCH and CORRAL codes, the Source Term Code Package, and the range of source terms from the uncertainty analysis in this study. The Surry plant was not one of the reference designs studied in the IDCOR program with the MAAP code. For the volatile radionuclides (the iodine, cesium, and tellurium groups), the principal difference between the Source Term Code Package and the earlier results is in the credit taken for retention within the reactor coolant system. In the uncertainty analysis, some of the credit taken in the Source Term Code Package for this retention is lost because of the potential for revaporization from reactor coolant system surfaces and the conversion of less volatile forms of iodine to more volatile forms during transport within the reactor coolant system. For the less volatile radionuclides, the uncertainty can either be in the direction of higher or lower releases than predicted by the Source Term Code Package, depending on the most important source of uncertainty.

Figure 5.3 shows a similar comparison for the interfacing-system LOCA sequence without water scrubbing in the safeguards building. For this sequence, an analysis was not actually performed in the Reactor Safety Study. The sequence was binned with the PWR2 bin, which is used to represent the Reactor Safety Study in the comparison. In the MARCH/CORRAL analysis, the only credit for retention of the volatile radionuclides was retention in the already failed safeguards building. It is not surprising that the MARCH/CORRAL release fractions for the volatile radionuclides are very high. The Source Term Code Package not only accounts for retention during transport in the reactor coolant system but also performs a more mechanistic analysis of aerosol retention processes in the safeguards building and in the containment after meltthrough of the reactor vessel. The principal sources of uncertainty in the analysis can lead to somewhat higher source terms for the volatile radionuclides, but the uncertainties are primarily in the direction of smaller source terms for the less volatile radionuclides.

The comparison of source terms from earlier studies with those of NUREG-1150 indicate that, at least for the sequences compared and considering uncertainties, the values in the Reactor Safety Study were not as conservative as has often been claimed. The presentation of source terms as a band of uncertainty provides much more insight into the state of knowledge than previous presentations of point estimates. It is not possible to make meaningful comparisons between source terms or to understand the significance of a source term in an absolute sense without doing so within the context of the associated uncertainties. If Figure 5.3 were considered without the NUREG-1150 results (with only the point-estimate values for the Reactor Safety Study, MARCH/CORRAL, and the Source Term Code Package), these points indicate a trend of a factor of two-to-ten reduction in source terms with the improvement in methods. When the uncertainty in the source term results is graphically displayed, however, it can be seen that the apparent decrease in source terms is within the uncertainty spread associated with outstanding phenomenological issues.

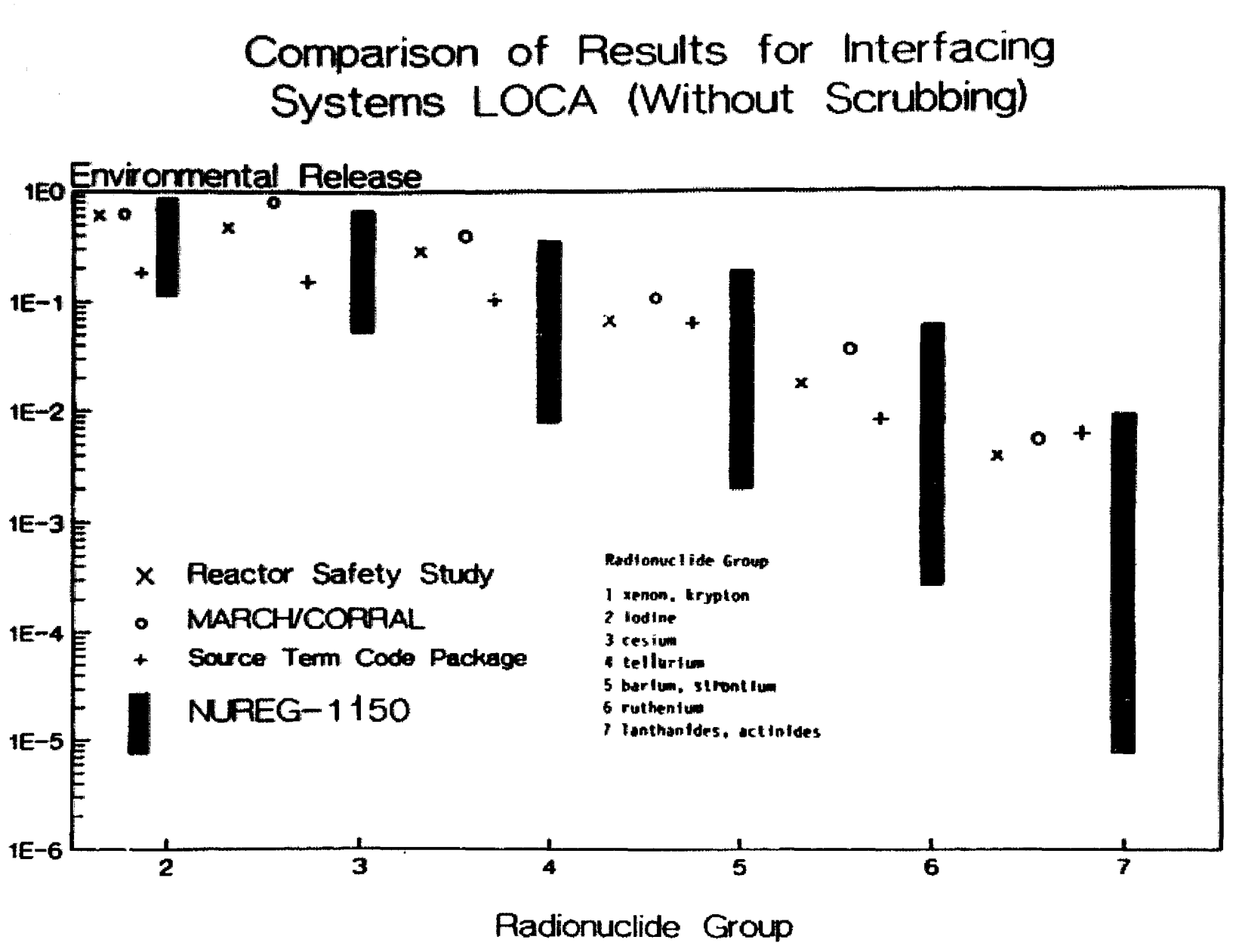


Figure 5.3 Comparison of results for interfacing-system LOCA at Surry

5.2.3 Plant-Specific Perspectives

Subsequent to the Three Mile Island accident, there has been considerable research undertaken to study the phenomena associated with the release and transport of radioactive material in severe accidents. A number of phenomena are treated in current models that have the potential to lead to greater retention of radionuclides within the plant in comparison with the Reactor Safety Study analyses. In particular, current models indicate that a significant fraction of the radionuclides released from fuel will deposit as aerosols, condense as vapors, or react as vapors with surfaces in the reactor coolant system. Similarly, the mechanistic treatment of aerosols in containment volumes in the Source Term Code Package predicts more retention of radioactive aerosols than the semiempirical model, CORRAL, used in the Reactor Safety Study and Reactor Safety Study Methodology Applications Program.

In comparing point estimates of radionuclide release results over the past decade there is an apparent trend with time toward smaller releases. When uncertainties are included in the comparison, however, the significance of the trend becomes less clear. The uncertainty bands associated with source terms are quite wide. For the Reactor Safety Study sequences, the Surry results fall within the band of uncertainty in NUREG-1150. This is because some of the phenomena identified in recent years have the potential to increase source terms rather than to decrease them. For example, the uncertainty associated with the revaporization of volatile radionuclides deposited on surfaces in the reactor coolant system tends to negate the credit taken for retention inside the reactor coolant system.

5.3 Results for Zion Nuclear Power Plant Unit 1

5.3.1 Ranges of Source Term Results

The same source term issues, levels, and weightings were considered in the uncertainty study for the Zion plant as in the stratified sampling study performed for the Surry plant (Ref. 5.17). Since it was decided that the break location in an interfacing-system LOCA sequence would be water covered, it was not necessary to consider the decontamination factor associated with water scrubbing.

As discussed in Section 4.3.2, the containment failure bins for the Zion plant were defined in almost an identical manner to those used for the Surry plant. The reactor coolant systems for the two plants are different; the Surry plant has three loops while the Zion plant has four loops. However, for a given plant damage state, the release of radionuclides from the fuel and transport within the reactor coolant system would be quite similar. There are also differences in the containment features between the two plants. Since for a given containment failure bin the mode and timing of failure are specified, the differences in containment features have a greater influence on the probability of a bin than on the release characteristics of a bin. It is not surprising, therefore, that the ranges of release fractions obtained for the Zion containment failure bins are very similar to those for Surry.

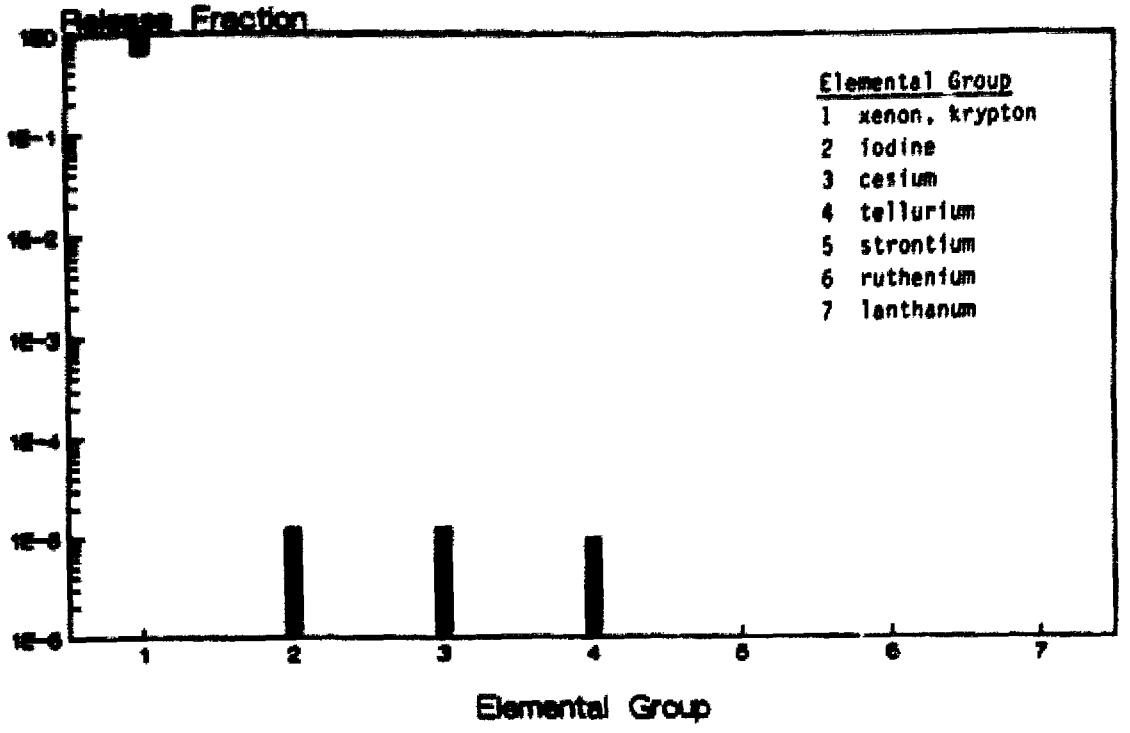
In the Surry analyses, the suite of codes that were later combined to form the Source Term Code Package were used in their stand-alone form (Ref. 5.18). For the Source Term Code Package, some improvements were made in the codes, including the treatment of release coefficients for the in-vessel analysis of release of radionuclides from fuel as a function of temperature. The most significant difference obtained was for the ruthenium group of radionuclides. As a result, for many of the bins the Zion values of ruthenium release are substantially lower than the Surry values. The only exceptions are the direct heating bins in which the magnitude of the ruthenium release is determined by the release during direct heating rather than by the in-vessel release period. The Source Term Code Package results also have two additional radionuclide groups. In order to better represent the chemical differences between elements, the barium/strontium group has been divided into two groups and the lanthanum group has been divided into two groups. For consistency with the Surry results, only seven groups are displayed. Results for the barium and cerium groups are quite similar to those obtained for the strontium and lanthanum groups, respectively.

Ranges of release fractions are illustrated in Figure 5.4 for four containment failure bins. Source Term Code Package results are shown on the figures for comparison. Bin 15, which involves no containment failure, has very small release fractions. The high release for the noble gases is somewhat misleading. Although it can be argued that the noble gases are not reactive and that they will all eventually be released, in actuality they would be largely decayed when released. Radioactive decay is accounted for in the consequence analysis codes. In the late failure case, Bin 8, in which containment sprays operate, the predicted release of iodine and cesium in the Source Term Code Package analysis is very small. In comparison, the NUREG-1150 uncertainty bands extend to release fractions of approximately 0.1. The important sources of uncertainty are the revolatilization of radionuclides from reactor coolant system surfaces and the late release of iodine. The high release assumptions used in the statistical sampling analysis for these two issues involve substantial fractions of the radionuclide inventories. Although the weights assigned to these levels were small, they were found to control the upper end of the iodine and cesium release terms.

5.3.2 Comparison With Other Studies

The Zion Probabilistic Safety Study (Ref. 5.19) used the Reactor Safety Study CORRAL code to estimate source terms for severe accident sequences. As a result, many of the mechanisms that are now recognized to be important in source term analysis were not considered.

The Zion plant was one of the reference plants in the IDCOR study (Ref. 5.20). Four scenarios were analyzed using the MAAP code. For the interfacing-system LOCA sequences, V, the predicted release fractions for the volatile radionuclides (iodine, cesium, and tellurium) were quite small (8×10^{-5} fraction of initial core inventory). The small release is primarily the result of extensive condensation of steam predicted in the auxiliary building in the MAAP analyses. A specific Source Term Code Package analysis was not performed for this case. The Brookhaven ranges of release fractions for the V plant damage state are substantially above this level of release, however, varying from



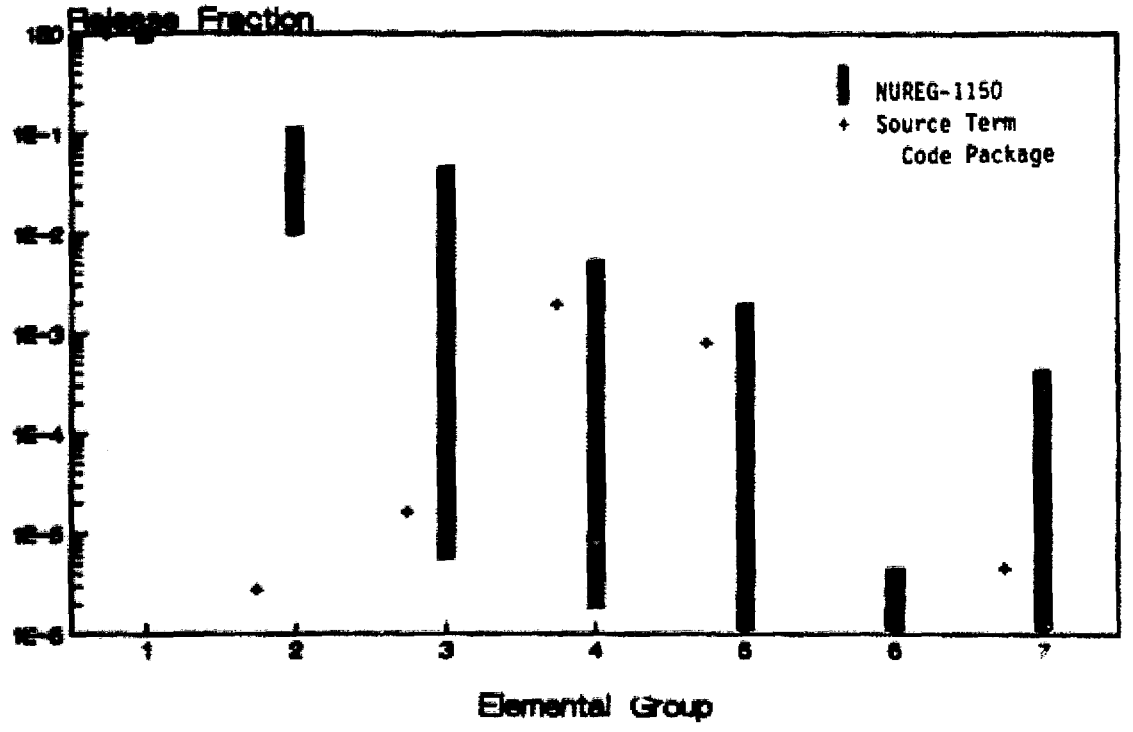


Figure 5.4 Ranges of release fractions for selected bins at Zion

10 percent to 80 percent for iodine, 5 percent to 70 percent for cesium, and 0.8 percent to 60 percent for tellurium.

Three variations of station blackout were also analyzed by IDCOR. The release fractions obtained for two of the cases with late containment failure were identical. Figure 5.5 compares the IDCOR values with the NUREG-1150 range for Bin 10, a late overpressure bin with leakage rather than rupture. Also shown in the figure are Source Term Code Package results. For each elemental group, the IDCOR results fall below or near the bottom of the NUREG-1150 uncertainty band.

A station blackout scenario was also analyzed by IDCOR in which there was assumed to be a preexisting breach in containment at the start of the accident. These results are compared with the range obtained for the corresponding Bin 1 in NUREG-1150 and with a Source Term Code Package run for a scenario within the bin. Again, the MAAP results are consistently below the NUREG-1150 range, indicative of technical disagreements in the source term models.

5.3.3 Plant-Specific Perspectives

The source term results for the Zion plant are quite similar to those obtained for the Surry plant and the plant-specific perspectives are the same. The uncertainties in the estimated source terms are quite large. The principal contributors to the uncertainties are basically the same as for the Surry plant. Comparisons made between the Source Term Code Package results and MAAP results indicated that the MAAP estimates for environmental release fractions were significantly smaller. It is very difficult to determine the precise source of the differences observed, however, without performing controlled comparisons for identical boundary conditions and input data.

5.4 Results for Sequoyah Nuclear Power Station Unit 1

5.4.1 Ranges of Source Term Results

The source term issues included in the statistical sampling analysis are very similar to those considered for the other PWR plants (Ref. 5.21). A few issues were added because of the unique aspects of the ice condenser design.

Decontamination Factor for Ice Condenser

Retention of radionuclides in the ice condenser region is a very important aspect limiting the release of radionuclides to the environment in this plant design. The effectiveness of the ice condenser is affected by the operability of the air-return fans. On the one hand, prior to containment failure the air-return fans can recycle the containment air through the ice condenser a number of times providing an opportunity for additional retention at each pass. By returning noncondensible gases to the lower compartment region, however, the air-return fans tend to reduce the effectiveness of the ice in retaining radionuclides for any single pass, since the decontamination factor is very sensitive to the fraction of steam in the flowing gas. Three boundary conditions were considered: (1) containment failure at vessel breach with the air-return fans operating, (2) containment failure prior to vessel breach with the air-return fans operating, and (3) containment failure at or before vessel breach with the air-return fans off.

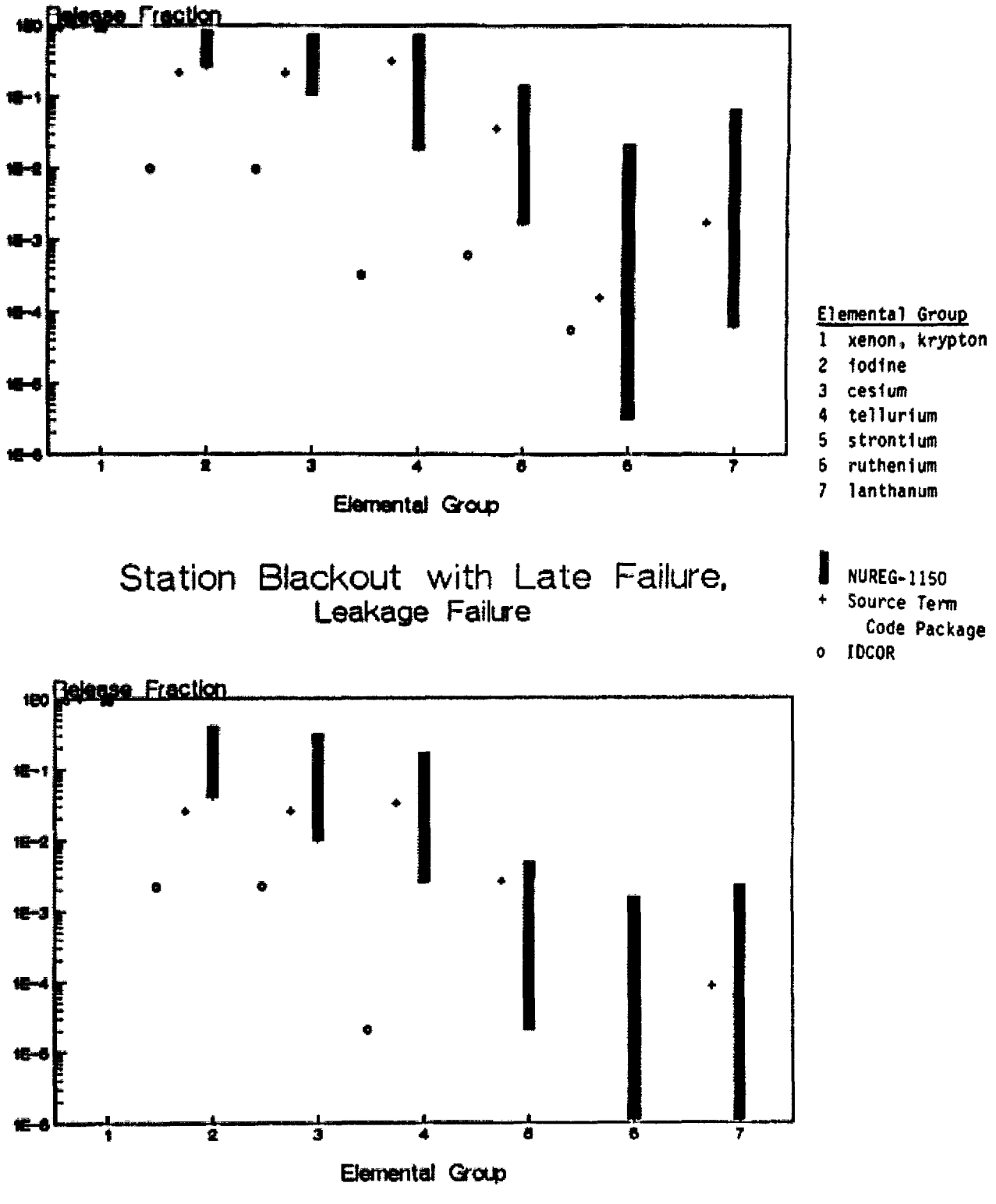


Figure 5.5 Comparison of results for station blackout scenarios at Zion

Scrubbing of Releases from Core-Concrete Interactions

As discussed previously, the potential exists in the Sequoyah design for a wide range of flooding conditions ranging from a dry cavity to a depth of 20 feet for complete injection of the refueling water storage tank and to higher values depending on the amount of ice melted. This issue is of most significance when the ice bed is depleted or bypassed at the time of core-concrete release.

Twenty-five containment failure bins were defined for the Sequoyah plant. The SIYYY (small break with failure of emergency core cooling in the recirculation mode) and SNNNY (failure of component cooling water system resulting in failure of the emergency core cooling, containment heat removal, and containment spray systems) stand out as the highest frequency plant damage states. For the SIYYY state, the most likely bins are Bin 23 in which there is no containment failure and Bin 21 in which the containment failure is late, the sprays operate, and the core concrete release is scrubbed by a deep pool of water. The range of release fractions for Bins 21 and 23 are illustrated in Figure 5.6. As expected, the release fractions for the no-containment-failure case (Bin 23) are quite small, even accounting for uncertainties. Most of the release fractions for Bin 21 are also small, except for the upper end of the uncertainty range for iodine release, which is driven by late iodine release uncertainties.

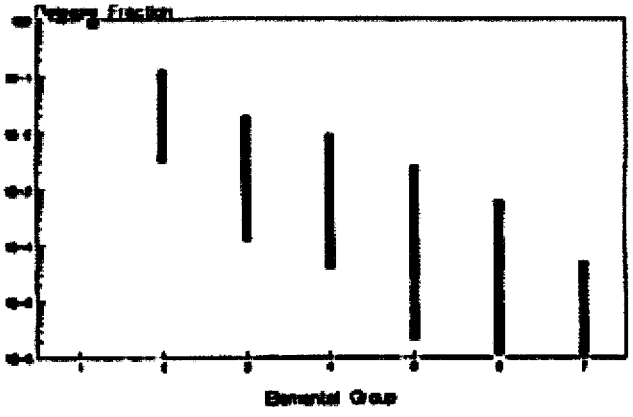
The SNNNY failure state would be expected to have more severe containment failure bins because fewer of the important containment safety systems are operational. However, because ac power is available and the igniters operate, the expected failure time is several hours after vessel breach. The principal bin for this plant damage state is Bin 19. The range of release fractions is shown in Figure 5.6.

Another important plant damage state is SNNNN, station blackout with failure of reactor coolant pump seals. The principal containment failure bins are Bin 1, in which there is early containment failure and the sprays and air-return fans are inoperable; Bin 21, which involves late containment failure as the result of hydrogen burning at the time ac power is restored; and Bin 19. A major difference between Bins 19 and 21 is that the containment spray system operates in Bin 21 after ac power is restored, removing aerosols from the containment atmosphere and providing water to the reactor cavity suppressing the core-concrete release.

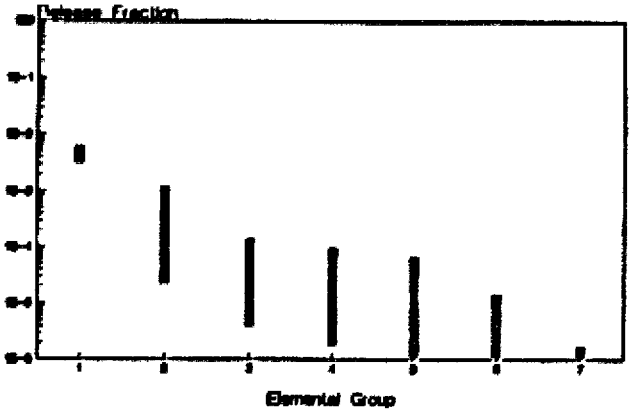
5.4.2 Comparison With Other Studies

Sequoyah was analyzed previously in the Reactor Safety Study Methodology Applications Program (Ref. 5.13) using the MARCH/CORRAL codes and, as one of the IDCOR reference plants, using MAAP. In the MARCH/CORRAL analyses the decontamination factor for the ice beds was input to the code. The base case value was a decontamination factor of 100. Some variations were performed using a value of 10. Mechanistic analyses using the ICEDF code (Ref. 5.22) in the Source Term Code Package indicate that a value as large as 100 is quite unlikely under the conditions anticipated. Figure 5.6 compares RSSMAP values with a decontamination factor of 10 with the NUREG-1150 range and with the results of the Source Term Code Package for Bin 1. The MARCH/CORRAL results tend to be near the upper end of the NUREG-1150 range. In contrast, the results obtained in the MARCH/CORRAL calculation with late containment failure and sprays operating, which

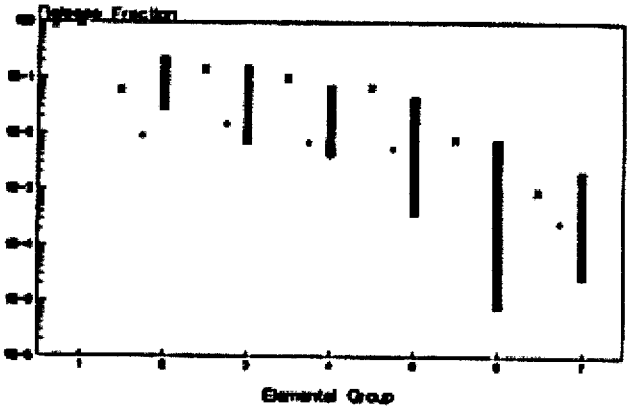
Bin 21, Late Failure
Sprays and Fans Operational



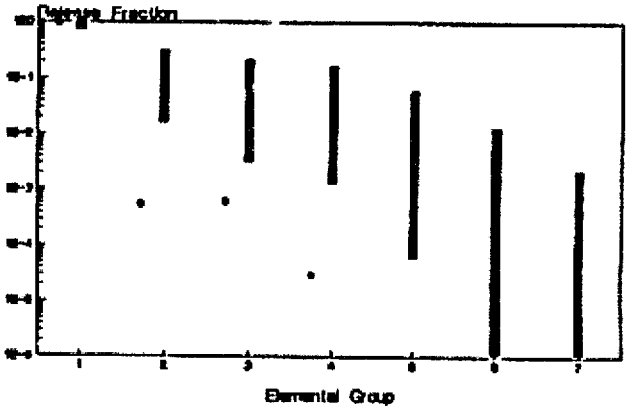
Bin 23, No Containment Failure



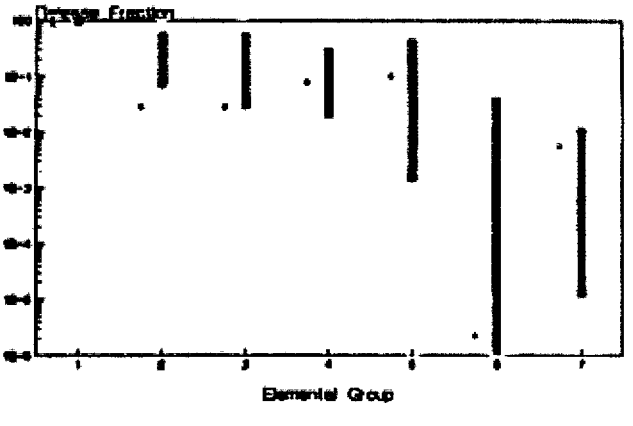
Bin 1 Early Failure, No AC Power
Ice Available at Vessel Breach



Bin 19, Late Failure,
Sprays and Fans Not Operational



Bin 3 Early Failure, No AC Power
Ice Bypassed



Elemental Group	
1	xenon, krypton
2	iodine
3	cesium
4	tellurium
5	strontium
6	ruthenium
7	lanthanum

▮	NUREG-1150
o	IDCOR
+	Source Term Code Package
X	RSSMAP

Figure 5.6 Ranges of release fractions for selected bins at Sequoyah

can be compared with Bin 21 results, are not shown in Figure 5.6 because the release was less than 10^{-6} for each elemental group of radionuclides.

In the IDCOR analyses, as discussed in Chapter 4, very few early containment failure cases were evaluated. One case that was analyzed is a small LOCA with failure of both the emergency core cooling system and the spray system in the recirculation mode with an impaired containment. Figure 5.7 provides a comparison of the IDCOR results with the Bin 15 range for the case of a large isolation failure. In this comparison, the IDCOR results fall within the NUREG-1150 uncertainty ranges. The IDCOR results for the iodine group fall at the bottom of the band because of the contribution of late release of iodine to the uncertainty in iodine release. The IDCOR results also fall near the bottom of the strontium band as the result of modeling differences related to core-concrete interaction release. A less favorable comparison is shown in Figure 5.6 for Bin 19. This is a late containment failure case involving station blackout conditions. The large difference in the results is more representative of differences in the modeling of containment behavior rather than differences in modeling source term behavior. In the IDCOR analyses, because of the smaller estimate of hydrogen production, the containment is predicted to remain intact for a day after vessel meltthrough. In the Sandia analyses, the failure time is only delayed a few hours. The nonvolatile groups have small release values in the IDCOR analysis (less than the 1×10^{-5} IDCOR cutoff). The NUREG-1150 bin involves quenching of the core debris in the reactor cavity and a very delayed period of core-concrete interaction after the water in the cavity has been boiled away.

5.4.2 Plant-Specific Perspectives

A number of features of the ice condenser design can play an important role in the mitigation of radionuclide release to the environment in a severe accident. The availability of the ice bed is the most important feature. As long as the ice bed is available at the time radionuclides are released, it is capable of reducing the release by an order of magnitude or more. A possible exception could be in the event of direct heating in which hot gases are transported too rapidly through the ice bed to allow effective decontamination.

Another important feature is the large depth of water that can develop in the reactor cavity. This water may prevent core-concrete interaction by forming a coolable debris bed, can decontaminate the core-concrete release if the debris is not coolable, may decrease the likelihood or effectiveness of direct heating after vessel failure, and can decontaminate the revolatilization release of volatile radionuclides from reactor coolant system surfaces.

The spray system provides another means for reducing the environmental release. If the containment remains intact for an extended period of time, the spray system can be particularly effective in the removal of suspended aerosols. In this regard, the operability of the hydrogen igniter system has an indirect but important impact on source terms because of its importance to the timing of containment failure.

For a number of bins the upper end of the band of uncertainty for the release of iodine is quite high. Uncertainties related to the long-term evolution of iodine from pools, organic iodine formation, and the revolatilization of iodine

Bin 15, Failure to Isolate

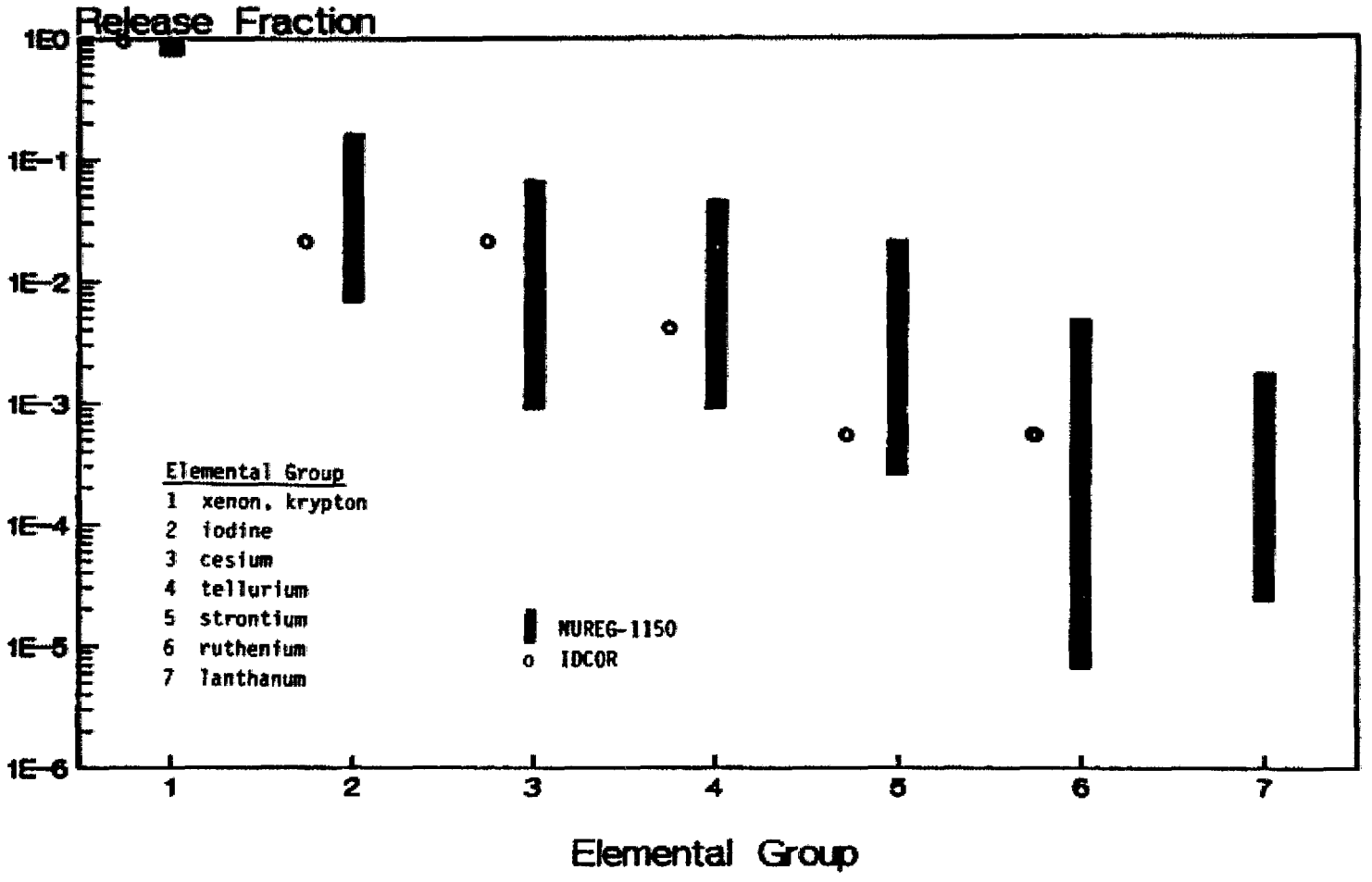


Figure 5.7 Comparison of results for failure to isolate containment at Sequoyah

from reactor coolant system surfaces are the key contributors to the top of the band. Ongoing research is investigating these issues; the final version of this report will reflect these new data.

5.5 Results for Peach Bottom Atomic Power Station Unit 2

5.5.1 Ranges of Source Term Results

The source term issues selected for the statistical sampling analyses for the two boiling water reactor plants were quite similar. For the Peach Bottom plant, nine issues were included in the analysis (Ref. 5.23):

1. Magnitude of in-vessel release from the fuel;

2. Amount of cesium iodide decomposition in the reactor pressure vessel;

3. The amount of radionuclide retention in the reactor coolant system;

4. Suppression pool decontamination factors for aerosols;

5. Suppression pool decontamination factors for volatile iodine;

6. Revolatilization of iodine and cesium from the reactor pressure vessel following vessel breach;

7. The magnitude of radionuclide release from the melt during core-concrete interactions;

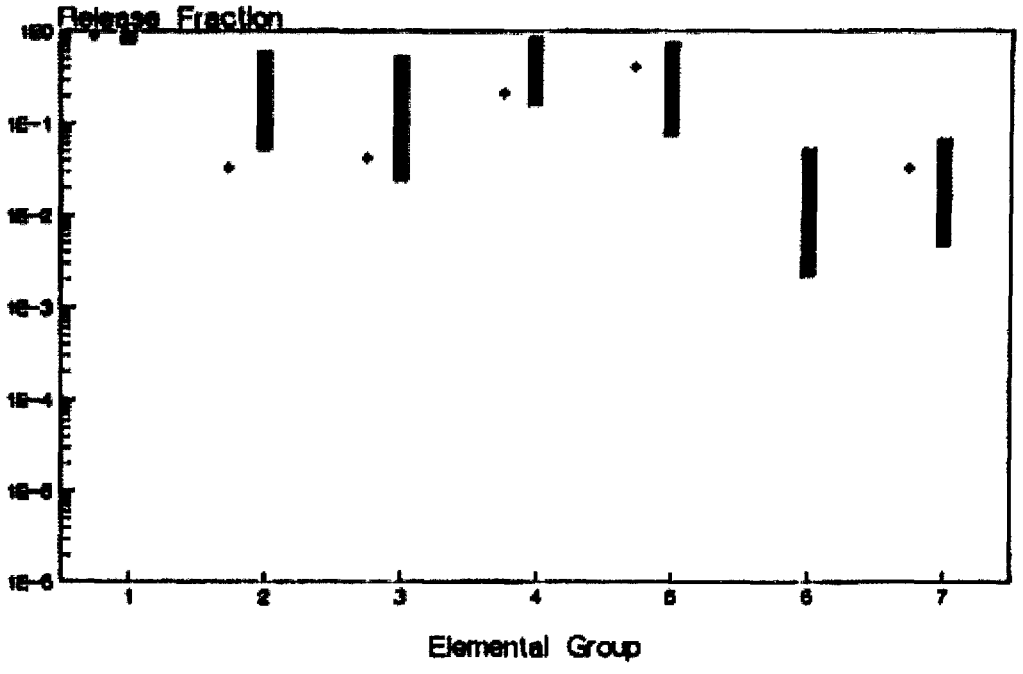
8. Reactor building and refueling bay decontamination factors; and

9. Late release of iodine from the pressure-suppression pool.

Figure 5.8 shows release fractions for two important Peach Bottom bins, designated Bin 7 and Bin 13 in Appendix E. Both are associated with station blackout scenarios leading to early core meltdown and early failure of the containment. In Bin 7 there is no direct heating. A moderate value of decontamination factor is applied to the reactor building. The strontium and lanthanum releases for the Source Term Code Package run are high relative to the NUREG-1150 uncertainty ranges because the amount of retention predicted for the reactor building was quite small. The predicted release of the ruthenium group was less than 10^{-6} of the core inventory and thus does not appear on the figure. In Bin 13 there is direct heating, and the reactor building decontamination factor is quite small. The results for the Source Term Code Package are shown for comparison. The code package values do not include the enhanced release associated with direct heating. In both comparisons, the NUREG-1150 uncertainty bands are higher for the iodine and cesium groups than the Source Term Code Package results because of the treatment of the late iodine and vessel revolatilization issues.

5.5.2 Comparison With Other Studies

The release fractions for some of the Reactor Safety Study scenarios were quite large. For example, the release fraction of iodine for the Reactor Safety Study BWR2 bin was 90 percent. The potential for reactor building retention



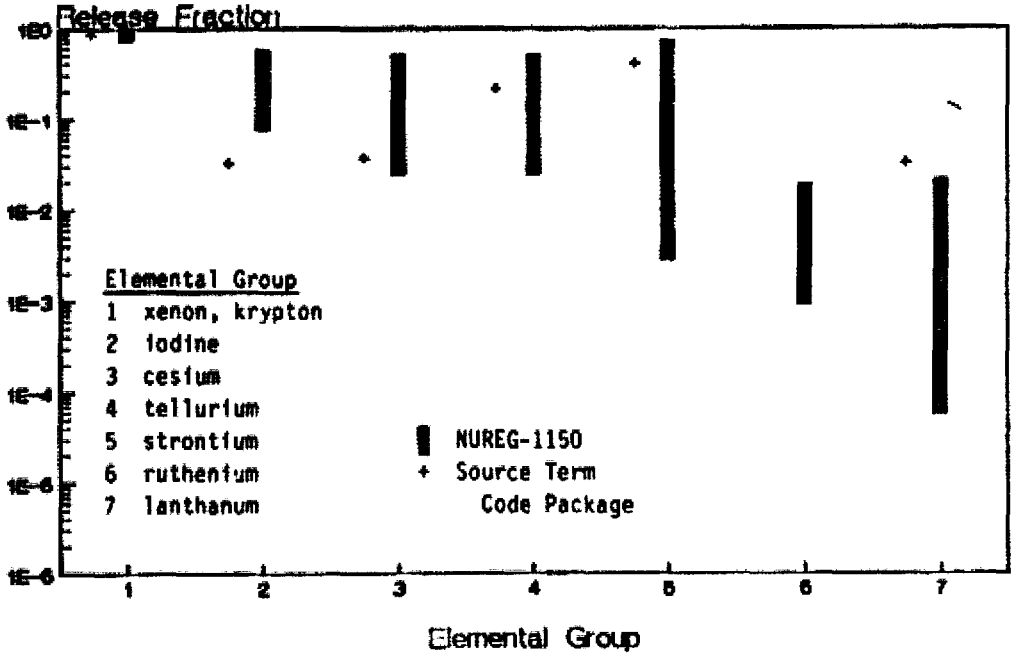


Figure 5.8 Ranges of release fractions for selected bins at Peach Bottom

was limited to deposition in the annulus between the steel containment shell and the reinforced concrete wall behind it. Although the station blackout sequence was not specifically analyzed in the Reactor Safety Study, a comparable sequence was analyzed with the MARCH/CORRAL code set as part of the rebaselining effort in the Reactor Safety Study Methodology Applications Program. In this accident, containment failure is predicted to follow vessel breach by approximately 45 minutes. Figure 5.9 provides a comparison of a variety of station blackout scenarios with delayed containment failure. In the Source Term Code Package analysis, the containment fails approximately 3 hours after vessel breach. In this case, a substantial amount of the strontium and lanthanum release from fuel during core-concrete attack occurs after containment failure. In comparison, for the NUREG-1150 range of releases shown for a late containment failure bin, the period of core-concrete release precedes containment failure. This is why the release fractions for the strontium and lanthanum groups are so much lower than the Source Term Code Package results.

MAAP results from the IDCOR program are also shown for comparison in Figure 5.9. In the IDCOR analysis, containment failure followed vessel breach by 6 hours. The Source Term Code Package and MAAP results are in reasonable agreement for the iodine, cesium, and tellurium groups. The MAAP releases of iodine and cesium are somewhat higher despite the longer time to containment failure because of the treatment of revolatilization from reactor coolant system surfaces in the analysis. The release of strontium in the Source Term Code Package analysis is orders of magnitude higher than the MAAP results because of the large release predicted by the Source Term Code Package during core-concrete interactions. The Source Term Code Package also predicts a large release of the lanthanum group, which the MAAP code did not model at the time of the reported analysis. The most severe scenarios analyzed in the IDCOR program were the transient with failure of long-term heat removal and an anticipated transient without scram, in both of which containment failure precedes core melting. In these cases, the predicted release of iodine to the environment was 20 percent and 10 percent, respectively.

5.5.3 Plant-Specific Perspectives

The source terms for the Peach Bottom plant are largely influenced by the mode and timing of containment failure. However, a number of other issues can also have a major influence. Since the likelihood of transient-initiated accidents is much greater than pipe break accidents leading to core melt, the release of radionuclides from the fuel that occurs in-vessel will initially be largely deposited on pressure vessel surfaces or captured by the suppression pool. As a result, the amount of the volatile radionuclides (iodine, cesium, and tellurium) that escape to the environment will be determined by subsequent phenomena such as the revolatilization from pressure vessel surfaces, late release of iodine from the suppression pool, and the amount of these elements still retained in the fuel at the time of vessel breach. Unfortunately, these issues all have large associated uncertainties.

Because the in-vessel release of radionuclides is likely to be substantially attenuated, the magnitude of the ex-vessel release of radionuclides in the Peach Bottom plant becomes that much more important. If early failure of containment occurs in the drywell, as expected in many scenarios in the Sandia study, only drywell sprays or deposition in the reactor building represents significant opportunities to decrease the release to the environment of the

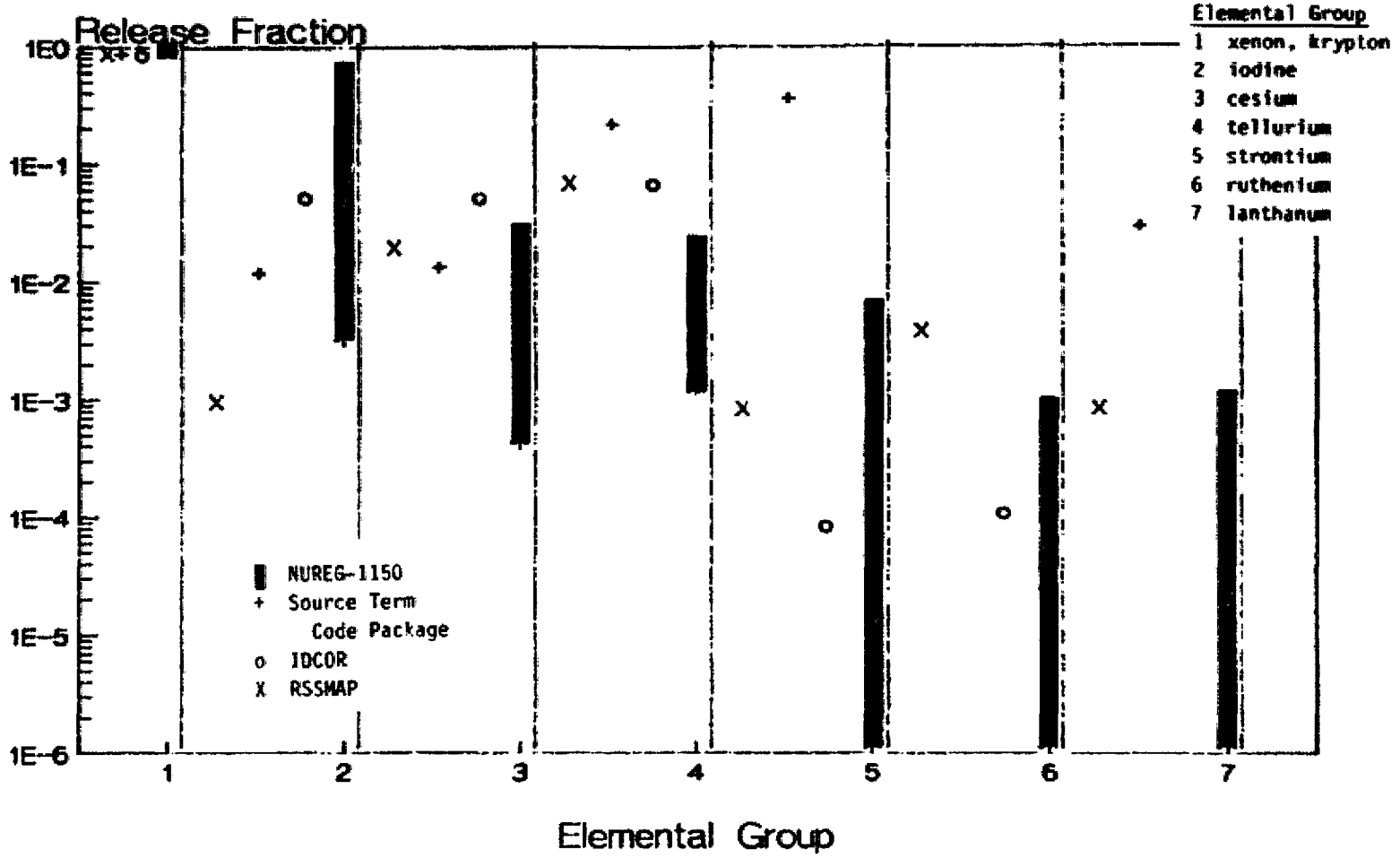


Figure 5.9 Comparison of results for station blackout scenarios at Peach Bottom

core-concrete release terms. Because of the high limestone composition of the aggregate in the Peach Bottom design, the CORCON/VANESA routines in the Source Term Code Package predict a particularly large release of radionuclides during core-concrete interactions. Whereas the consequences of most severe accident scenarios in pressurized water reactors tend to be controlled by the quantities of iodine and cesium released, in the Peach Bottom plant the lanthanum, cerium, barium, and strontium groups can have a greater radiological impact than the more volatile groups.

5.6 Results for Grand Gulf Nuclear Station Unit 1

5.6.1 Ranges of Source Term Results

Only four source term issues were included in the statistical sampling analysis for the Grand Gulf facility (Ref. 5.24). The selection was based on an assessment of the importance of different source term issues in the Peach Bottom analysis.

1. Revolatilization of iodine and cesium from the reactor pressure vessel following vessel breach.

2. The magnitude of radionuclide release from the melt during core-concrete interactions.

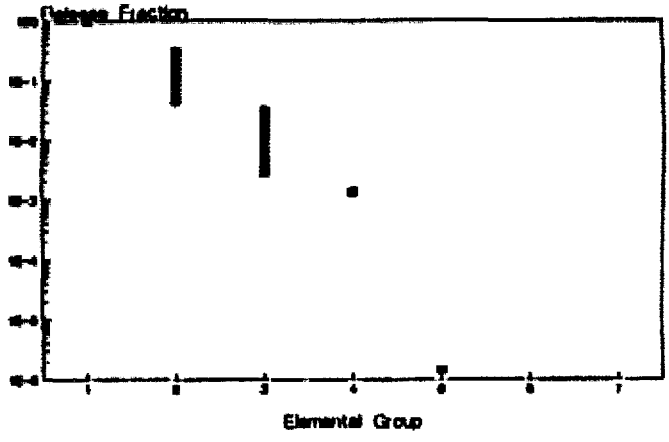
3. Scrubbing of core-concrete interactions by overlying water in the drywell.

4. Late release of iodine from the pressure-suppression pool.

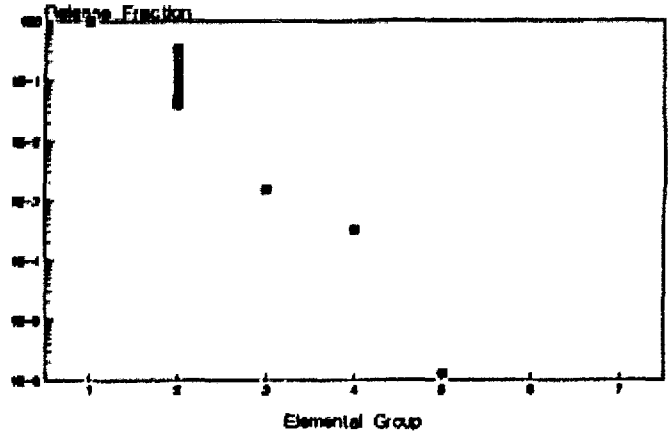
Because of the large number of containment failure bins, no single bin completely dominates any of the measures of risk. However, a few bins can be identified that are particularly important contributors. The principal contributors to early failure probability are Bins 137, 138, 144, and 145, which are variations of a single scenario. Each involves early failure of the containment, no containment sprays, no core-concrete interaction, and nominal leakage from the drywell. These conditions are expected in station blackout scenarios with recovery of ac power. In Bins 137 and 144, containment failure precedes core melting and in Bins 138 and 145 containment failure occurs at the end of the in-vessel melting period. Bins 137 and 138 are complete core meltdown scenarios involving meltthrough of the pressure vessel, whereas in Bins 144 and 145 emergency core cooling is recovered and the melting is arrested within the vessel. As indicated in Figure 5.10, the release fractions for Bins 137 and 138 are identical as are the release fractions for Bins 144 and 145. The timing of containment failure and the energy release vary as associated with the relative timing of core melting and containment failure. Since there is no ex-vessel release or it is effectively scrubbed for all of these bins, the very limited statistical sampling treatment for Grand Gulf only identified uncertainty ranges for iodine and cesium. As expected, however, the iodine and cesium releases for these scenarios are by far the most important among the different elemental groups.

The most important source of uncertainty is in the late release of iodine from the suppression pool. In the statistical sampling treatment, up to 50 percent

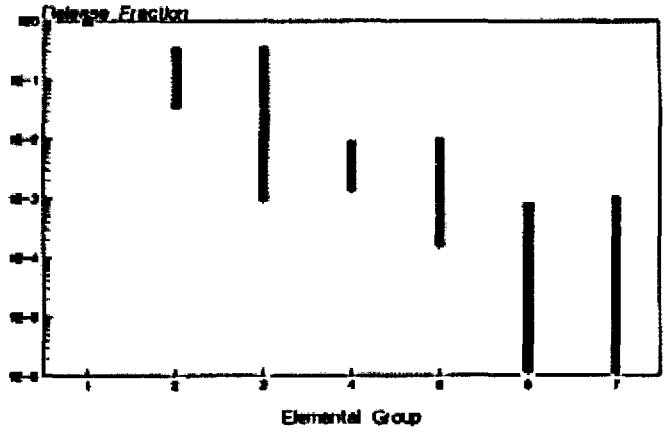
Bin 137/138 Early Failure, No Sprays,
Nominal Drywell Leakage



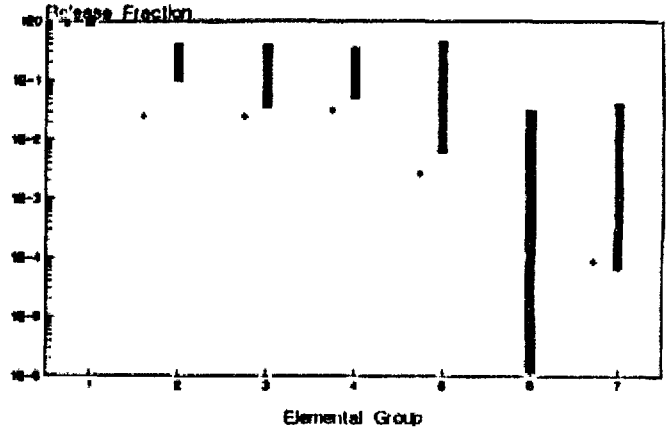
Bin 144/145 Early Failure, No Sprays,
Nominal Drywell Leakage, Recovery



Bin 132 Late Failure, Damaged Drywell



Bin 128 Early Failure, Damaged Drywell



Elemental Group
1 xenon, krypton
2 iodine
3 cesium
4 tellurium
5 strontium
6 ruthenium
7 lanthanum

▮ NUREG-1150
\+ Source Term Code Package

Figure 5.10 Ranges of release fractions for selected bins at Grand Gulf

of the iodine captured in the pool could potentially be released from containment. Although the probability assigned to this amount of release was not high, the resulting release is quite large. This uncertainty affects nearly all the Grand Gulf release bins leading to an upper bound on the release uncertainty of approximately one-third of the core inventory of iodine. As discussed in Chapter 11, this issue is intended to be the subject of additional study prior to the publication of the final version of this report.

Another important contributor to risk in Grand Gulf is Bin 128, in which there is early failure of the containment accompanied by damage to the drywell resulting in significant bypass of the pool. In this scenario, the core-concrete interaction is delayed until water in the reactor cavity has been boiled away. As indicated in Figure 5.10, the uncertainties in the release of the nonvolatile radionuclides are quite broad with the potential for large releases of the lanthanum and strontium groups, as well as for the more volatile radionuclides.

The Source Term Code Package results fall slightly below the uncertainty ranges for Bin 128. In the sequence analyzed, core-concrete interactions began before the water in the pedestal region had been boiled away. As a result, the releases for the nonvolatile radionuclide groups were partially scrubbed in the Source Term Code Package analysis. The releases of the volatile radionuclide groups, iodine and cesium, are also lower for the Source Term Code Package calculation because of the late release mechanisms accounted for in the NUREG-1150 uncertainty analysis.

Some late containment failure scenarios can also result in potentially large releases. Bin 132 involves late containment failure but with a damaged drywell. As a result, the cesium, which after some period of delay is revolatilized from the pressure vessel surfaces, is able to bypass the suppression pool and escape from the containment. More recent analytical work at Sandia indicates that the potential for revaporization from surfaces is not as great as simple analyses based on the vapor pressures of pure substances predict. These additional data will be reflected in the final version of this report.

5.6.2 Comparison With Other Studies

Neither the IDCOR nor the Reactor Safety Study Methodology Applications Program analyzed station blackout scenarios. The methodology of the Reactor Safety Study Methodology Applications Program was quite outdated in comparison with modern analytical tools. The CORRAL code was used to analyze the transport of radionuclides in the containment. No retention of radionuclides within the reactor coolant system was taken into account. A decontamination factor for the suppression pool was input to the code but a value of unity (no retention) was assumed for cases in which the pool was hot. As a result, the predicted release fractions for many of the sequences analyzed were quite high.

Figure 5.11 shows a comparison between IDCOR, Source Term Code Package, and the NUREG-1150 range of release fractions for an anticipated transient without scram scenario in which containment failure precedes core meltdown. Except for the ruthenium release group, the IDCOR results fall beneath the lower boundary of the NUREG-1150 uncertainty band. Release estimates for all the IDCOR scenarios analyzed are quite low because it was assumed that there was minimal bypass of the suppression pool and because the potential for significant vaporization of iodine from the pool was not considered possible.

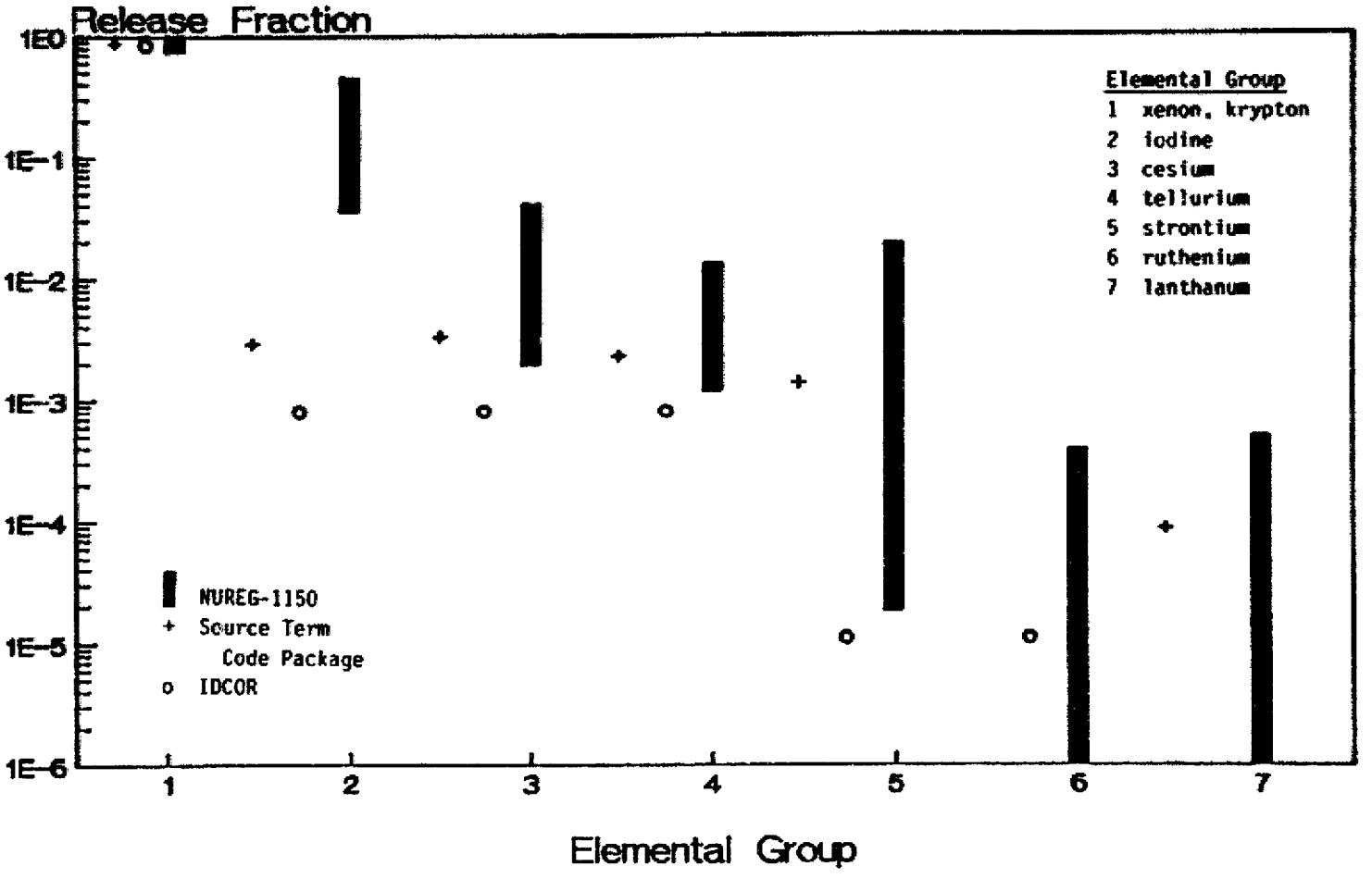


Figure 5.11 Comparison of results for anticipated transient without scram scenario at Grand Gulf

5.6.3 Plant-Specific Perspectives

In general, the source terms for the Grand Gulf plant are lower than for the other plants analyzed. A handful of uncertainty issues do lead to large release estimates for the upper bound of the uncertainty estimate for some containment failure bins. The most important issue appears to be the late release of iodine from the suppression pool. Although the potential for large release of iodine must be recognized with the current state of understanding, it is expected that ongoing research will lead to a narrowing of the uncertainty. Similarly, a large potential for the release of iodine and cesium from reactor coolant system surfaces is recognized in the statistical sampling analysis. This is another area where ongoing research should lead to a reduction in uncertainties in the future. As noted previously, the final version of this report will reflect more recent data in these areas.

There is little question at this time that the suppression pool would capture a significant fraction of the radionuclides released in a severe accident in the Mark III design, as long as the suppression pool is not bypassed. The degree of decontamination would depend on the flow rate of gas entering the pool, the fraction of noncondensible gas, the temperature of the pool, and the characteristics of the aerosols and vapors entering the pool. However, because of the depth of the vents and spargers, the uncertainty assessment indicated that the level of decontamination would be expected to be at least a factor of three to ten and would most likely be larger. The amount of the iodine captured by the pool that would subsequently be released is a major source of uncertainty, however.

5.7 Perspectives

Magnitude of Uncertainties

> The magnitude of the uncertainties in the prediction of severe accident phenomena and source terms is large.

The ranges of environmental release terms obtained for the containment failure bins in the NUREG-1150 analyses are quite broad. For the volatile groups of radionuclides, the ranges are typically one to two orders of magnitude and, for the more refractory groups of radionuclides, two to three orders of magnitude. These ranges are similar to those obtained in the QUEST program (Ref. 5.6), which was undertaken to assess the uncertainties associated with the NUREG-0956 suite of codes. The NRC is currently supporting additional work at Brookhaven National Laboratory in the QUASAR program (Ref. 5.25) and at Sandia National Laboratories in the PRUEP program to further characterize the uncertainties in source term analyses.

The specific source term issues that have the greatest impact on the uncertainties in source terms depend not only on the design of the plant but also on the specific plant damage state. A number of source term issues were important contributors to the uncertainties in the predicted release terms.

1. Late release of iodine was a particularly large contributor to the release of the iodine group for the Grand Gulf and Peach Bottom plants. Although

most of the iodine was predicted to be captured on pressure vessel surfaces or in the suppression pool initially, the source term specialists believed there was the potential for the subsequent release of a large fraction of the captured iodine.

2. Reevolution of volatile radionuclides from reactor coolant system surfaces also led to high upper levels on the release fractions for the volatile groups for both the PWRs and the BWRs.

3. The magnitude of the core-concrete release was a key source of uncertainty for all plants. In the BWRs, the in-vessel release is typically largely captured by the suppression pool or is initially retained on reactor vessel surfaces. In the Peach Bottom plant, the ex-vessel release, on the other hand, was often predicted to occur after containment failure, with bypass of the suppression pool. Similarly, in the Grand Gulf design large releases are typically only predicted to occur in conjunction with drywell failure, again with the core-concrete release bypassing the suppression pool.

Because there is significant uncertainty in the analysis of severe accident source terms, different analysts may obtain substantially different source terms for nominally the same accident scenarios. In comparing IDCOR source term results with NUREG-1150 uncertainty bands, there is an obvious tendency for the IDCOR release fractions to be lower than the NRC contractor results. As discussed throughout this chapter (and in Appendix L), there has been a general tendency in the MAAP analysis to assume somewhat more optimistic outcomes of certain phenomenological issues such as extent of hydrogen generation. It should also be recognized, however, that the NRC contractors have included a wide range of outcomes, including potentially pessimistic values for some issues in the NUREG-1150 statistical sampling analyses.

The magnitude of the environmental release fractions is not the only important characteristic of the source term. The timing of release can also be very important, particularly with regard to the production of early radiological health effects. For comparable accident scenarios, the IDCOR analyses typically predict later times of containment failure than the Source Term Code Package. In addition, the largest IDCOR release fractions occur as the result of the revolatilization of iodine, cesium, and tellurium from reactor coolant system surfaces over an extended period of time. The effect of release timing on consequences is illustrated in Chapter 6, "Offsite Consequence Analysis."

Improvement in Understanding

Substantial improvements have been made in the understanding of severe accident processes and the confidence with which source terms can be estimated.

In this study, care has been taken to display the assessed uncertainties associated with the analysis of accident source terms. A major shortcoming of the Reactor Safety Study was the limited treatment that was made of the uncertainties in the estimation of branching probabilities on the containment event tree and in the calculation of severe accident source terms. To a great degree, the ability of the Reactor Safety Study analysts to estimate the uncertainties

in severe accident processes was limited by the state of the art. Many of the severe accident issues that now head the list of areas requiring resolution were completely unknown to the Reactor Safety Study analysts a decade ago. In the intervening years, particularly subsequent to the Three Mile Island accident, there have been major experimental and code development efforts that have explored every aspect of severe accident behavior to some extent. Although there remain large uncertainties in the quantitative estimation of severe accident source terms, there is also a high level of confidence that the qualitative understanding of severe accident behavior that has been developed is correct. Further research will undoubtedly yield surprises, but there should not be big surprises that lead to major revisions in basic perceptions of severe accident mechanisms.

Severe accident phenomena are very complex. Although significant progress has been made in the past few years in developing a better understanding of these processes, some aspects of severe accidents are very difficult and expensive to simulate experimentally. This is particularly true of the behavior of core melt progression after the normal fuel pin geometry of the core is destroyed. To the extent that source terms depend on predicting this type of behavior, it may not be reasonable to expect that the computer codes will be validated with a high degree of confidence. Significant reductions in the source term uncertainty bands are possible. However, the reduction of the range of uncertainty below approximately an order of magnitude is probably not achievable with reasonable allocation of research funds.

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ATTACHMENT 10

Excerpts from NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" (Dec. 1990)

NUREG–1150
Vol. 1

Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants

Final Summary Report

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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NUREG–1150
Vol. 1

Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants

Final Summary Report

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Division of Systems Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

ABSTRACT

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

This report was first published in February 1987 as a draft for public comment. Extensive peer review and public comment were received. As a result, both the underlying technical analyses and the report itself were substantially changed. A second version of the report was published in June 1989 as a draft for peer review. Two peer reviews of the second version were performed. One was sponsored by NRC; its results are published as the NRC report NUREG–1420. A second was sponsored by the American Nuclear Society (ANS); its report has also been completed and is available from the ANS. The comments by both groups were generally positive and recommended that a final version of the report be published as soon as practical and without performing any major reanalysis. With this direction, the NRC proceeded to generate this final version of the report.

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

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PART I

Introduction

Summary of Methods

1. INTRODUCTION

1.1 Background

In 1975, the U.S. Nuclear Regulatory Commission (NRC) completed the first study of the probabilities and consequences of severe reactor accidents in commercial nuclear power plants—the Reactor Safety Study (RSS) (Ref. 1.1). This work for the first time used the techniques of probabilistic risk analysis (PRA) for the study of core meltdown accidents in two commercial nuclear power plants. The RSS indicated that the probabilities of such accidents were higher than previously believed but that the offsite consequences were significantly lower. The product of probability and consequence—a measure of the risk of severe accidents—was estimated to be quite low relative to other man-made and naturally occurring risks.

Following the completion of these first PRAs, the NRC initiated research programs to improve the staff's ability to assess the risks of severe accidents in light-water reactors. Development began on advanced methods for assessing the frequencies of accidents. Improved means for the collection and use of plant operational data were put into place, and advanced methods for assessing the impacts of human errors and other common-cause failures were developed. In addition, research was begun on key severe accident physical processes identified in the RSS, such as the interactions of molten core material with concrete.

In parallel, the NRC staff began to gradually introduce the use of PRA in its regulatory process. The importance to public risk of a spectrum of generic safety issues facing the staff was investigated and a list of higher priority issues developed (Ref. 1.2). Risk studies of other plant designs were begun (Ref. 1.3). However, such uses of PRA by the staff were significantly tempered by the peer review of the RSS, commonly known as the Lewis Committee report (Ref. 1.4), and the subsequent Commission policy guidance to the staff (Ref. 1.5).

The 1979 accident at Three Mile Island substantially changed the character of NRC's analysis of severe accidents and its use of PRA. Based on the comments and recommendations of both major investigations of this accident (the Kemeny and Rogovin studies (Refs. 1.6 and 1.7)), a substantial research program on severe accident phenomenology was planned and initiated (Refs. 1.8 and 1.9). This program included experimental and analytical studies of accident physical processes. Computer models were developed to simulate these processes. The Kemeny and Rogovin investigations also recommended that PRA be used more by the staff to complement its traditional, nonprobabilistic methods of analyzing nuclear plant safety. In addition, the Rogovin investigation recommended that NRC policy on severe accidents be reconsidered in two respects: the need to specifically consider more severe accidents (e.g., those involving multiple system failures) in the licensing process, and the need for probabilistic safety goals to help define the level of plant safety that was "safe enough."

By the mid-1980's, the technology for analyzing the physical processes of severe accidents had evolved to the point that a new computational model of severe accident physical processes had been developed—the Source Term Code Package—and subjected to peer review (Ref. 1.10). General procedures for performing PRAs were developed (Ref. 1.11), and a summary of PRA perspectives available at that time was published (Ref. 1.12). The Commission had developed and approved policy guidance on how severe accident risks were to be assessed by NRC (Ref. 1.13) as well as safety goals against which these risks could be measured (Ref. 1.14) and methods by which potential safety improvements could be evaluated (Ref. 1.15).

In 1988, the staff requested information on the assessment of severe accident vulnerabilities by each licensed nuclear power plant (Ref. 1.16). This "individual plant examination" could be done either with PRA or other approved means. (In response, virtually all licensees indicated that they intended to perform PRAs in their assessments.) The staff also developed its plans for integrating the reviews of these examinations with other severe accident-related activities by the staff and for coming to closure on severe accident issues on the set of operating nuclear power plants (Ref. 1.17).

One principal supporting element to the staff's severe accident closure process is the reassessment of the risks of such accidents, using the technology developed through the 1980's. This reassessment updates the first staff PRA—the Reactor Safety Study—and provides a "snapshot" (in time) of estimated plant risks in 1988 for five commercial nuclear power plants of different design. For this reassessment, the plants have been studied by teams of PRA specialists under contract to NRC (Refs. 1.18 through 1.31). This report,

NUREG–1150, summarizes the results of these studies and provides perspectives on how the results may be used by the NRC staff in carrying out its safety and regulatory responsibilities.

NUREG–1150 was first issued in draft form in February 1987 for public comment. In response, 55 sets of comments were received, totaling approximately 800 pages. In addition, comments were received from three organized peer review committees, two sponsored by NRC (Refs. 1.32 and 1.33) and one by the American Nuclear Society (Ref. 1.34). Appendix D provides a summary of the principal comments (and their authors) on this first draft of NUREG–1150 and the staff's responses. A second draft version of NUREG–1150 was issued in June 1989, taking into account the comments received and reflecting improvements in methods identified in the course of performing the draft risk analyses, in the design and operation of the studied plants, and in the information base of severe accident phenomenology.

Because of the significant criticisms of the first draft of NUREG–1150, and the substantial changes made in response, the second version of the report was issued as a draft for peer review. A review committee was established under the provisions of the Federal Advisory Committee Act (Ref. 1.35). This committee reviewed the report for approximately 1 year and published its results in August 1990 (Ref. 1.36). In parallel, the American Nuclear Society-sponsored review of the report continued; its results were published in June 1990 (Ref. 1.37). Also, the NRC's Advisory Committee on Reactor Safeguards (ACRS) reviewed the analyses and provided comments (Ref. 1.38). Four sets of public comments were also received. While all committees suggested that some changes be made to the report, the comments received were, in general, positive, with all review committees recommending that the report be published in final form as soon as possible and without extensive reanalysis or changes.

This is the final version of NUREG–1150. In keeping with the review committees' recommendations, the staff has made relatively modest changes to the second draft of the report, with essentially no additional technical analysis. (Appendix E provides a summary of the comments and recommendations made by the review committees and the staff's responses. It also includes the ACRS comments in toto.)

Two other recommendations of the review committees should also be noted here. First, the ANS committee indicated that the changes made between the first and second drafts of NUREG–1150 were so substantial that the former should be considered, in effect, obsolete. The staff agrees with this comment and recommends that the analyses and results contained in the first draft no longer be used. Second, the ACRS cautioned that the results should be used only by those who have a thorough understanding of their limitations. The staff agrees with this comment as well.

1.2 Objectives

The objectives of this report are:

- To provide a current assessment of the severe accident risks of five nuclear power plants of different design, which:
  - Provides a snapshot of risks reflecting plant design and operational characteristics, related failure data, and severe accident phenomenological information available as of March 1988;
  - Updates the estimates of NRC's 1975 risk assessment, the Reactor Safety Study;
  - Includes quantitative estimates of risk uncertainty in response to a principal criticism of the Reactor Safety Study; and
  - Identifies plant-specific risk vulnerabilities for the five studied plants, supporting the development of the NRC's individual plant examination (IPE) process;
- To summarize the perspectives gained in performing these risk analyses, with respect to:
  - Issues significant to severe accident frequencies, containment performance, and risks;
  - Risk-significant uncertainties that may merit further research;
  - Comparisons with NRC's safety goals; and
  - The potential benefits of a severe accident management program in reducing accident frequencies; and
- To provide a set of PRA models and results that can support the ongoing prioritization of potential safety issues and related research.

In considering these objectives and the risk analyses in this and supporting contractor reports, it is important to consider both what NUREG–1150 is and what it is not:

- NUREG–1150 is a snapshot in time of severe accident risks in five specific commercial nuclear power plants. This snapshot is obtained using, in general, PRA techniques and severe accident phenomenological information of the mid-1980's, but with significant advances in certain areas. The plant analyses reflect design and operational information as of roughly March 1988.

- NUREG–1150 is an important resource document for the NRC staff, providing quantitative and qualitative PRA information on a set of five commercial nuclear power plants of different design with respect to important severe accident sequences, and a means for investigating where safety improvements might best be pursued, the cost-effectiveness of possible plant modifications, the importance of generic safety issues, and the sensitivity of risks to issues as they arise.

- NUREG–1150 is an estimate of the actual risks of the five studied plants. It is a set of modern PRAs, having the limitations of all such studies. These limitations relate to the quantitative measurement of certain types of human actions (errors of commission, heroic recovery actions); variations in the licensee's organizational/management safety commitments; failure rates of equipment, especially to common-cause effects such as maintenance, environment, design and construction errors, and aging; sabotage risks; and an incomplete understanding of the physical progression and consequences of core damage accidents.

- NUREG–1150 is **not** the sole basis for making plant-specific or generic regulatory decisions. Such decisions must be more broadly based on information on the extant set of regulatory requirements, reflecting the present level of required safety, cost-benefit studies (in some circumstances), risk analysis results (from this and other relevant PRAs), and other technical and legal considerations.

- NUREG–1150 is **not** an estimate of the risks of all commercial nuclear power plants in the United States or abroad. One of the clear perspectives from this study of severe accident risks and other such studies is that characteristics of design and operation specific to individual plants can have a substantial impact on the estimated risks.

1.3 Scope of Risk Analyses

The five risk analyses discussed in this report include the analysis of the frequency of severe accidents, the performance of containment and other mitigative systems and structures in such accidents, and the offsite consequences (health effects, property damage, etc.) of these accidents. In assessing accident frequencies, the five risk analyses consider events initiated while the reactor is at full-power operation.* For two plants, both "internal" events (e.g., random failures of plant equipment, operator errors) and "external" events (e.g., earthquakes, fires) have been considered as initiating events. For the remaining three plants, only internal events have been studied.

The five commercial nuclear power plants studied in this report are:

- Unit 1 of the Surry Power Station, a Westinghouse-designed three-loop reactor in a subatmospheric containment building, located near Williamsburg, Virginia (including the analysis of both internal and external events);**

- Unit 1 of the Zion Nuclear Plant, a Westinghouse-designed four-loop reactor in a large, dry containment building, located near Chicago, Illinois;

- Unit 1 of the Sequoyah Nuclear Power Plant, a Westinghouse-designed four-loop reactor in an ice condenser containment building, located near Chattanooga, Tennessee;

- Unit 2 of the Peach Bottom Atomic Power Station, a General Electric-designed BWR-4 reactor in a Mark I containment building, located near Lancaster, Pennsylvania (including the analysis of both internal and external events);** and

- Unit 1 of the Grand Gulf Nuclear Station, a General Electric-designed BWR-6 reactor in a Mark III containment building, located near Vicksburg, Mississippi.

*Analysis of shutdown and low-power accident risks for the Surry and Grand Gulf plants was initiated in FY 1989.

**These plants were used as models in the Reactor Safety Study.

The external-event analysis summarized in this report includes discussion of the core damage frequency and containment performance from seismically initiated accidents. The offsite consequences and risks are not provided. The reason for this limitation is related to the offsite effects of a large earthquake.

Two sets of hazard curves are used (and reported separately) in the seismic analysis. One set was prepared by Lawrence Livermore National Laboratory (Ref. 1.39) under contract to NRC. Analysis performed using these hazard curves (which have been prepared for the Surry and Peach Bottom sites and other reactor sites east of the Rocky Mountains) suggest that relatively rare but large earthquakes contribute significantly to the risk from seismic events. A second set of hazard curves was also prepared for sites east of the Rocky Mountains for the Electric Power Research Institute (Ref. 1.40). Although both projects made extensive use of expert judgment and formal methods for obtaining these judgments (as did many parts of this project, as discussed in Chapter 2), there were some important differences in methods. Nonetheless, the NRC believes that at present both methods are fundamentally sound.

A significant portion of the estimated seismic-induced core damage frequency for the Surry and Peach Bottom plants arises from large earthquakes. Should such a large earthquake occur in the Eastern United States (e.g., at the Surry or Peach Bottom site), there would likely be substantial damage to some older residential structures, commercial structures, and high hazard facilities such as dams. This could have a major societal impact over a large region, including property damage, injuries, and fatalities. The technology for assessing losses from such earthquakes is a developing one. There are several studies of this technology at this time, including work at the United States Geological Survey. There is no agreed-upon method for this purpose, although a recent report of the National Academy of Sciences (Ref. 1.41) suggests some broad guidelines. The NRC, in its promulgation of safety goals, indicated a preference for quantitative goals in the form of a ratio or percentage of nuclear risks relative to non-nuclear risks. For example, the probability of an early fatality from a nuclear power plant accident should not exceed 1/1000 of the "background" accidental death rate. The NRC intends to further investigate the methods for assessing losses from earthquakes in the vicinity of the Surry and Peach Bottom sites with a view of comparing the ratio of seismically induced reactor accident losses with the overall losses. There has been at least one study (Ref. 1.42) that suggests that the reactor accident contribution to seismic losses is very small relative to the non-nuclear losses. However, this study did not explicitly consider the two sites of interest in this report.

In contrast, because they are aimed at experts in the field of risk analysis, the contractor reports underlying this report (Refs. 1.20, 1.21, 1.27, and 1.28) present the seismic risk results in the form of a set of sensitivity analyses. These analyses consider the effects of the alternative sets of earthquake frequencies and severities noted above, as well as alternative assumptions on the performance of containment structures in large earthquakes, and the possible regional effects of earthquakes (lack of shelter, difficulty in evacuation and relocation, nonradiologically induced injuries and fatalities, etc.) on estimates of plant risk. The reader is cautioned that the results presented in the contractor reports should be used only in the broader context of the overall societal response.

1.4 Structure of NUREG–1150 and Supporting Documents

This report has three parts:

- Part I discusses the background, objectives, and methods used in this assessment of severe accident risks;
- Part II provides summary results and discussion of the individual risk studies of the five examined plants; and
- Part III provides:
  - Perspectives on the collective results of these five PRAs, organized by the principal subject areas of risk analysis: accident frequencies; accident progression, containment loadings, and structural response; transport of radioactive material; offsite consequences; and integrated risk (the product of frequencies and consequences);
  - Discussion of how the risk estimates have changed (and reasons why) for the two plants studied in both the Reactor Safety Study and this report (Surry and Peach Bottom); and

- Discussion of the role of NUREG–1150 as a resource document in the staff's assessment of severe accidents.

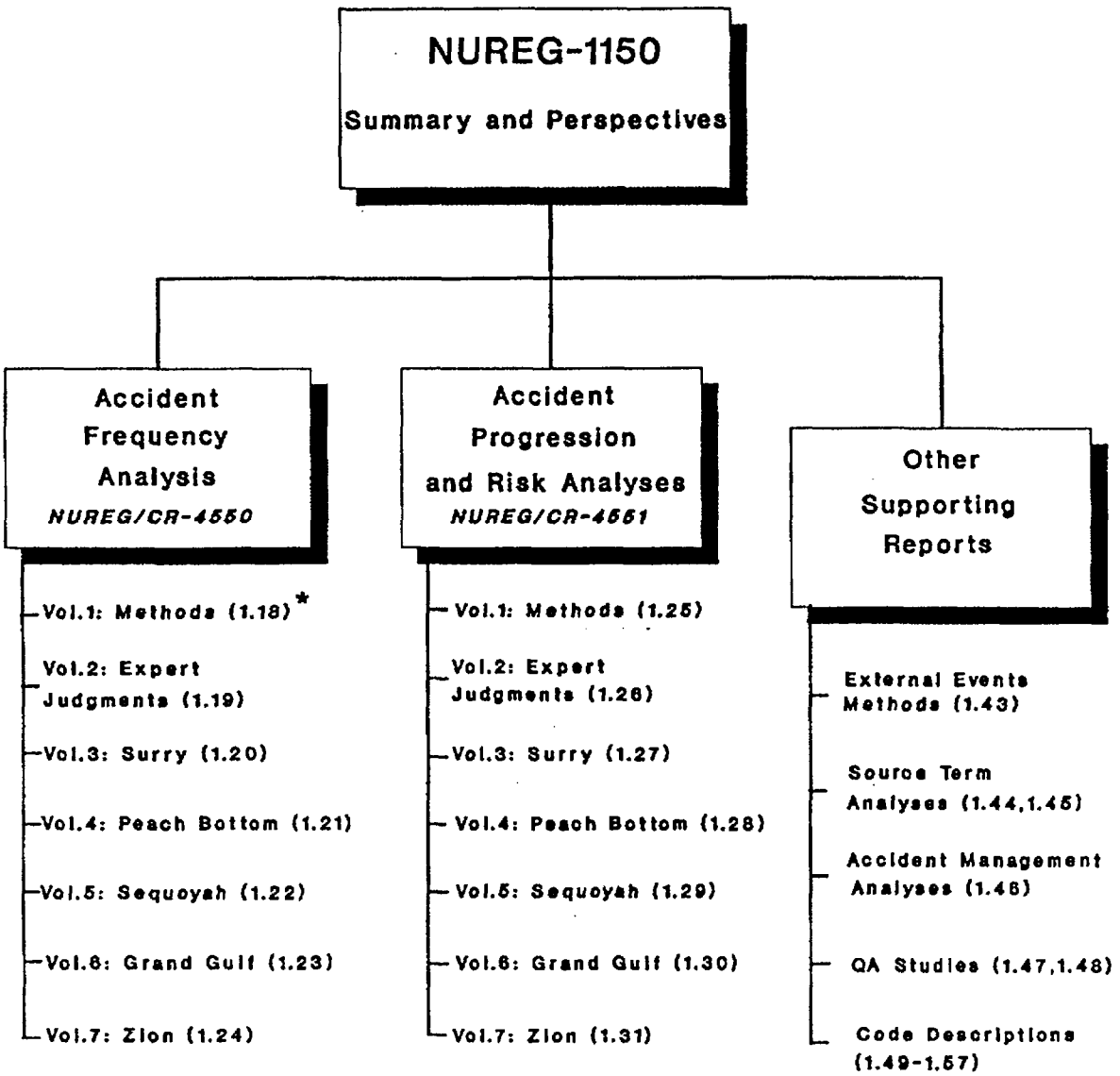
Three appendices are contained in Volume 2 of this report. Appendix A discusses in greater detail the methods used to perform the five risk analyses.* In Appendix B, an example calculation is provided to describe the flow of data through the individual elements of the NUREG–1150 risk analysis process. Appendix C provides supplemental information on key technical issues in the risk analyses. Volume 3 contains two additional appendices. As indicated previously, Appendices D and E provide summaries of comments received on the first and second versions of draft NUREG–1150, respectively, and the associated responses.

As noted above, this report provides a summary of five PRAs performed under contract to NRC. Volume 1 is written for an intended audience of people with a general familiarity with nuclear reactor safety and probabilistic risk analysis. Appendices A, B, and C are written for an intended audience of specialists in reactor safety and risk analysis.

As shown in Figure 1.1, supporting this report are a series of contractor reports providing the detailed substance of the five risk studies. These reports are written for specialists in reactor safety and PRA. The staff's principal contractors for this work have been:

- Sandia National Laboratories, Albuquerque, New Mexico;
- Brookhaven National Laboratory, Upton, New York;
- Idaho National Engineering Laboratory, Idaho Falls, Idaho;
- Battelle Memorial Institute, Columbus, Ohio; and
- Los Alamos Scientific Laboratory, Los Alamos, New Mexico.

*The sections of Appendix A are adapted, with editorial modification, from References 1.18 and 1.25.



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2. SUMMARY OF METHODS

2.1 Introduction

In many respects, the five probabilistic risk analyses (PRAs) performed in support of this report (Refs. 2.1 through 2.14) have been performed using PRA methods typical of the mid-1980's (Refs. 2.15 and 2.16). However, in certain areas, more advanced techniques have been applied. In particular, advancements have occurred in the following areas:

- The estimation of the size of the uncertainties in core damage frequency* and risk due to incomplete understanding of the systems responses, severe accident progression, containment building structural response, and in-plant radioactive material transport;
- The formal elicitation and documentation of expert judgments;**
- The more detailed definition of plant damage states, improving the efficiency of the interface between the accident frequency and accident progression analyses;
- The types of events and outcomes explicitly considered in the accident progression and containment loading analyses;
- The analysis of radioactive material releases and the integration of experimental and calculational results into this analysis;
- The use of more efficient methods for estimating the frequency of core damage accidents resulting from external events (e.g., earthquakes); and
- The application of new computer models in the analysis and integration of risk information.

The assessment of severe accident risks performed for this report can be divided into five general parts (shown in Fig. 2.1): accident frequency; accident progression, containment loading, and structural response; transport of radioactive material; offsite consequences; and integrated risk analyses. This last part combines the information from the first four parts into estimates of risk. These parts are described in Sections 2.2, 2.3, 2.4, 2.5, and 2.8, respectively. Additional discussion of each of these parts is provided in Appendix A and in substantial detail in References 2.1 and 2.8.

Because the estimation of uncertainties in core damage frequency and risk due to uncertainties in the constituent analyses is important to the overall objectives of this study, the descriptions of the constituent analyses will include discussions of uncertainties. The parts of the accident frequency analyses, the accident progression analyses, the containment building structural response analyses, and the radioactive transport analyses that are highly uncertain have been identified. In place of single "best estimates" for parameters representing these uncertain parts of the analyses, probability distributions have been developed. The methods for obtaining probability distributions for uncertain parameters (through, for the most part, the use of expert judgment) and the methods by which the probability distributions in the constituent analyses are propagated through the analyses to yield estimates of the uncertainties in core damage frequency and risk are described in Sections 2.7 and 2.6, respectively. Additional discussion of these two subjects is provided in Sections 6 and 7 of Appendix A and in detail in References 2.1 and 2.8.

The principal results obtained from the five PRAs that form the basis of this report are probability distributions. For simplicity, these distributions may be described by a number of statistical characteristics. The characteristics generally used in this report are the mean, the median, and 5th percentile and 95th percentile of the distributions. No one characteristic conveys all the information necessary to describe the distribution, and any one can be misleading. In particular, for very broad distributions (spanning several orders of magnitude), the mean can be dominated by the high value part of the distribution. If this is also a low probability part of the distribution, the estimate of the mean can exhibit a high degree of statistical variability. Conclusions based on mean values of such distributions must be carefully examined to ensure that dependencies and trends seen in the mean values apply to entire distributions. Conclusions stated in this report have not been based entirely on characteristics of mean values. In some circumstances, median values or entire distributions are used. In particular, the

*Table 2.1 provides definitions of key terms used in this report.

**Risk analyses and other technical studies routinely make use of expert judgment. It is the use of formal procedures to obtain and document these judgments that is noteworthy here.

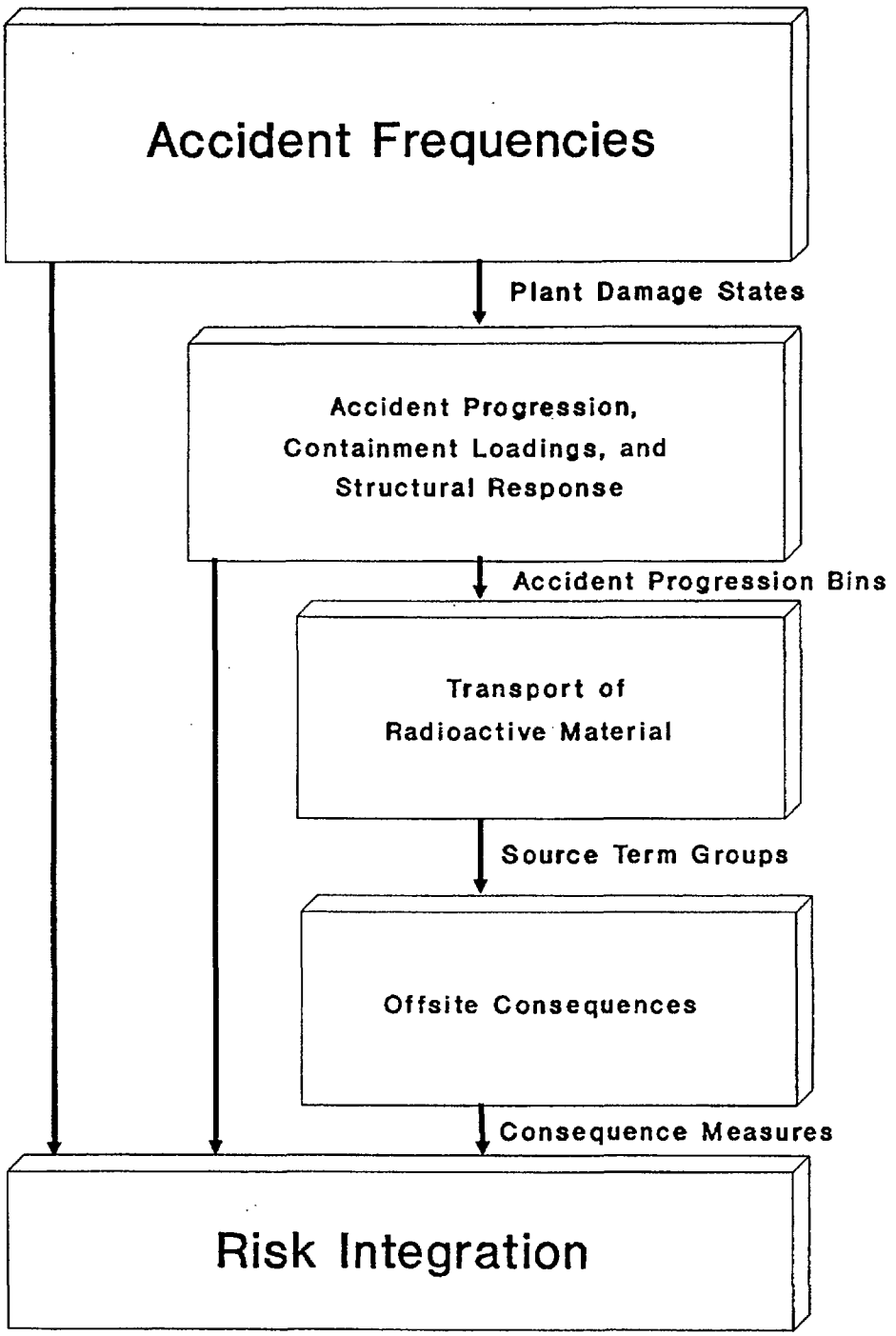


Figure 2.1 Elements of risk analysis process.

Table 2.1 Definition of some key NUREG–1150 risk analysis terms.

Core Damage Frequency: The frequency of combinations of initiating events, hardware failures, and human errors leading to core uncovery with reflooding of the core not imminently expected. For the pressurized water reactors (PWRs) discussed in this report, it was assumed that onset of core damage occurs at uncovery of the top of the active fuel (without imminent recovery). For the boiling water reactors (BWRs) discussed in this report, it was assumed that onset of core damage would occur when the water level was less than 2 feet above the bottom of the active fuel (without imminent recovery). (Ref. 2.1 discusses the reasons for the BWR/PWR differences.)

Internal Initiating Events: Initiating events (e.g., transient events requiring reactor shutdown, pipe breaks) occurring during the normal power generation of a nuclear power plant. In keeping with PRA tradition, loss of offsite power is considered an internal initiating event.

External Initiating Events: Events occurring away from the reactor site that result in initiating events in the plant. In keeping with PRA tradition, some events occurring within the plant during normal power plant operation, e.g., fires and floods initiated within the plant, are included in this category.

Plant Damage State: A group of accident sequences that has similar characteristics with respect to accident progression and containment engineered safety feature operability.*

Accident Progression Bin: A group of postulated accidents that has similar characteristics with respect to (for this summary report) the timing of containment building failure and other factors that determine the amount of radioactive material released.* These are analogous to containment failure modes used in previous PRAs.

Early Containment Failure: Those containment failures occurring before or within a few minutes of reactor vessel breach for PWRs and those failures occurring before or within 2 hours of vessel breach for BWRs. Containment bypass failures (e.g., interfacing-system loss-of-coolant accidents) are categorized separately from early failures.

Source Term: The fractions defining the portion of the radionuclide inventory in the reactor at the start of an accident that is released to the environment. Also included in the source term are the initial elevation, energy, and timing of the release.

Source Term Group: A group of releases of radioactive material that has similar characteristics with respect to the potential for causing early and latent cancer fatality consequences and warning times.

Offsite Consequences: The effects of a release of radioactive material from the power plant site, measured (for this summary report) as the number of early fatalities in the area surrounding the site and within 1 mile of the site boundary, latent cancer fatalities in the area surrounding the site and within 10 miles of the power plant, and population dose in the area surrounding the site and within 50 miles of the power plant.

Probability Density Function: The derivative of the cumulative distribution function. A function used to calculate the probability that a random variable (e.g., amount of hydrogen generated in a severe accident) will fall in a given interval. That probability is proportional to the height of the distribution function in the given interval.

Cumulative Distribution Function: The cumulative distribution function gives the probability of a parameter being less than or equal to a specified value. The *complementary cumulative distribution function* gives the probability of a parameter value being equal to or greater than a specified value.

*Groupings of this sort can be made in a variety of ways; the contractor reports underlying this report provide more detailed groups (Refs. 2.3 through 2.7 and 2.10 through 2.14).

reader is cautioned that an estimated mean may vary by about a factor of two because of sample variation. This variation can also impact the relative contribution of factors (e.g., plant damage states) to the mean (particularly small contributions).

In many risk analyses, "best estimate" analyses are performed. For these studies, many input parameters, even highly uncertain ones, are represented by single "best" values rather than probability distributions as done in this study. The resulting estimate of risk calculated with such best estimate parameter values is not simply related to the mean, median, or any other value of the distributions of risk calculated in this study.

As is implicit in Figure 2.1, the five principal risk analysis parts have clearly defined interfaces through which summary information passes to and from the constituent parts of the analysis and which provide convenient intermediate results for examination and review. Such summary information will be provided in this report; the form of the information presented will be described in the following sections.

2.2 Accident Frequency Estimation

The accident frequency estimation methods underlying this report considered accidents initiated by events occurring during the normal full-power generation* of a nuclear power plant ("internal events") and those initiated by events occurring away from the plant site ("external events"). (Historically, accidents initiated by loss of offsite power have been included in the category of internal events, while fires and floods within the plant during normal operation have been included in the category of external events. This tradition is continued in this report.) The discussion below summarizes accident frequency estimation methods first for internally initiated accidents, followed by those for externally initiated accidents.

2.2.1 Methods

2.2.1.1 Internal-Event Methods

The first part of the analysis shown in Figure 2.1 ("Accident Frequencies") represents the estimation of the frequencies of accident sequences leading to core damage. In this portion of the analysis, combinations of potential accident initiating events (e.g., a pipe break in the reactor coolant system) and system failures that could result in core damage are defined and frequencies of occurrence calculated. The methods for performing this analysis are discussed in Appendix A and in considerable detail in Reference 2.1. In summary, the basic steps in this analysis are:

- *Plant Familiarization*: In this step, information is assembled from plant documentation using such sources as the Final Safety Analysis Report, piping and instrumentation diagrams, technical specifications, operating procedures, and maintenance records, as well as a plant site visit to inspect the facility, gather further data, and clarify information with plant personnel. Regular contact is maintained with the plant personnel throughout the study to ensure that current information is used. The analyses discussed in this report reflect each plant's status as of approximately March 1988. This step of the accident frequency analysis was performed in a manner typical of recent PRAs (e.g., as described in Ref. 2.15).

- *Accident Sequence Initiating Event Analysis*: Information is assembled on the types of accident initiating events of potential interest for the specific plant. The initiating events identified include those that could result from support system failures, such as electric power or cooling water faults. Frequencies of initiating events are then assessed. In some cases, the assessed frequencies of certain events were very low; such events were not carried forward into the remaining analysis. Then, the safety functions required to prevent core damage for the individual initiating events are identified, along with specific plant systems required to perform those safety functions, the systems' success criteria (e.g., how much water flow is required from a pumping system), and related operating procedures. The initiating events are then grouped based upon the similarity of response needed from the various plant systems. This step of the analysis was performed in a manner typical of recent PRAs.

- *Accident Sequence Event Tree Analysis*: Using information from the previous step, system event trees that display the combinations of plant system failures that can result in core damage are constructed for each initiating event group. An individual path through such an event tree (an accident sequence) identifies specific combinations of system successes and failures leading to (or avoiding) core damage. As such, the event tree qualitatively identifies what systems must fail in a plant in order to cause core damage (the associated

*Accidents initiated in non-full-power operation are the subject of ongoing study for the Surry and Grand Gulf plants.

system failure probabilities are obtained in following steps). This step of the analysis was performed in a more advanced manner relative to other recent PRAs. For example, the analyses supporting this report considered a significantly greater number of systems in the event trees, including the potential effects on core damage processes from failures of containment functions and systems.

- *Systems Analysis*: In order to estimate the frequencies of accident sequences, the failure probability of each system must be obtained. The important contributors to failure of each system are defined using fault tree analysis methods. Such methods allow the analyst to identify the ways in which system failure may occur, assign failure probabilities to individual plant components (e.g., pumps or valves) and human actions related to the system's operation, and combine the failure probabilities of individual components into an overall system failure probability. This step was performed in a manner typical of that of recent PRAs. The level of detail was determined by the system's relative importance to core damage frequency, based on screening assessments and perspectives from other studies and PRAs.*

- *Dependent and Subtle Failure Analysis*: In addition to the combining of individual component failures, plant systems can fail as a result of the failure of multiple components due to a common cause. Such "dependent failures" may be separated into two types. First, there are direct functional dependencies that can lead to failure of multiple components (e.g., lack of electric power from emergency diesel generators causing failure of emergency core cooling systems). Such dependencies are incorporated directly into the fault or event trees. Second, there are dependent failures that have been experienced in plant operations due to less direct causes and often for which no direct causal relationships have been found. Various methods exist for incorporating such "miscellaneous" failures into the quantification of system fault trees. For this study, a modified "beta factor" method was used (Ref. 2.17). This step of the accident frequency analysis was performed in greater depth than that of typical recent PRAs, in that considerable effort was devoted to generating beta factors for multiple failures (i.e., more than two) using recent advances in common-cause analytical methods. In addition, a subtle failure "checklist" was developed and used. This checklist defined subtle failures found in previous PRAs.

- *Human Reliability Analysis*: As noted in previous steps, explicit consideration of human error was included in the analysis. Errors of two types were incorporated: pre-accident errors, including, for example, failure to properly return equipment to service after maintenance; and post-accident initiation errors, including failure to properly diagnose or respond to and recover from accident conditions. In order to assess failure probabilities for such events, operating procedures for the specific plant under study were obtained and reviewed. In general, the analysis of such errors was made using methods typical of recent PRAs (i.e., modifications of the "THERP" method (Ref. 2.18)) but at a somewhat reduced level of effort. An initial screening analysis was performed to focus the analysis to the potentially most important operator actions (including recovery actions), permitting some savings of effort. More detailed analyses were performed for the BWR anticipated transient without scram (ATWS) accident sequences (Refs. 2.6 and 2.19).

- *Data Base Analysis*: In general, a common data base of equipment and human failure rates and initiating event frequencies was used in the five plant risk analyses, based on operating experience in all commercial nuclear power plants (Ref. 2.1). In addition, the operating experience of each plant studied for this report was examined for relevant failure data on key systems and equipment. The "generic" data base (from all plants) was then replaced with plant-specific data (if available) for these key components in cases where the plant-specific data were significantly different. The methods used to obtain and apply plant-specific data were typical of those of recent PRAs; however, the level of effort expended was less than that generally performed because of limitations in the original analysis scope and, in some cases, because a plant's operating life had been too short to generate an adequate data base.

- *Accident Sequence Quantification Analysis*: In this step, the information from the

*The reader is cautioned that the level of analysis detail and screening assessments used for systems in this study was based on the designs of each of the plants. Thus, it should not be inferred that the results of such assessments necessarily apply to other plants.

preceding steps was assembled into an assessment of the frequencies of individual accident sequences, using the fault trees and event trees to combine probabilities of individual events. This was performed in a manner typical of recent PRAs.

- *Plant Damage State Analysis*: In order to assist the analysis of the physical processes of core damage accidents (i.e., the subsequent steps in a risk analysis), it is convenient to group the various combinations of events comprising the accident sequences into "plant damage states." These states are defined by the operability of plant systems (e.g., the availability of containment spray systems) and by certain key physical conditions in an accident (e.g., reactor coolant system pressure). The definition of the plant damage states and the associated frequencies are the principal products provided to the next step in the risk analysis, i.e., the analysis of accident progression, containment loadings, and structural response. This step was performed in a manner more advanced than most recent PRAs because of the complexity of the interface with the more detailed accident progression analysis.

- *Uncertainty Analysis and Expert Judgment*: As noted in Section 2.1, the risk analyses underlying this report include the quantitative analysis of uncertainties. This analysis was performed using the Latin hypercube sampling technique (Ref. 2.20), a specialized modification of Monte Carlo simulation techniques often used in the combination of uncertainties. The elicitation of expert judgments was necessary to develop the probability distributions for some individual parameters in this uncertainty analysis. For certain key issues in the uncertainty analysis, panels of experts were convened to discuss and help develop the needed probability distributions. The methods used for uncertainty analysis and expert judgment elicitation are discussed in Sections 2.6 and 2.7. For the accident frequency analysis, six issues were evaluated by two expert panels and probability distributions developed; these issues are shown in Table 2.2. Probability distributions were developed for many other parameters as well. Section C. 1 of Appendix C includes a listing of the set of accident frequency issues assigned distributions for the Surry plant. Similar lists for the other plants may be found in References 2.11 through 2.14.

Appendix B provides a detailed example calculation for a particular accident (a station blackout) at the Surry plant. Section B.2 of that appendix describes the analysis of the accident sequence frequency.

It should be noted that the methods used in the accident frequency analysis of the Zion plant varied from those described above. A PRA was completed for this plant by the licensee (Commonwealth Edison Company) in 1981 (Ref. 2.21). This PRA was subsequently reviewed by the NRC staff and its contractors (Ref. 2.22), with the review completed in 1985. For the Zion accident

Table 2.2 Accident frequency analysis issues evaluated by expert panels.

• Accident Frequency Analysis Panel
Failure probabilities for check valves in the quantification of interfacing-system LOCA frequencies (PWRs)
Physical effects of containment structural or vent failures on core cooling equipment (BWRs)
Innovative recovery actions in long-term accident sequences (PWRs and BWRs)
Pipe rupture frequency in component cooling water system (Zion)
Use of high-pressure service water system as source for drywell sprays (Peach Bottom)
• Reactor Coolant Pump Seal Performance Panel
Frequency and size of reactor coolant pump seal failures (PWRs)

frequency analysis summarized in this report, this previous PRA (as modified by the 1985 staff review) was updated to reflect the plant design and operational features in place in early 1988. As such, the Zion accident frequency analysis relied substantially on the previous PRA, rather than performing a new study.

The methods used to perform the Zion accident frequency analysis are discussed in greater detail in Section A.2.2 of Appendix A and in Reference 2.7.*

2.2.1.2 External-Event Methods

The analysis of accident frequencies for the Surry and Peach Bottom plants included the consideration of accidents initiated by external events (e.g., earthquakes, floods, fires) (Refs. 2.3 and 2.4). The methods used to perform these analyses are more efficient versions of previous methods and are described in Section A.2.3 of Appendix A and in more detail in Reference 2.23.

1. External-Event Methods: Seismic Analysis

The seismic analysis methods performed for this study consisted of seven steps. Briefly, these are:

- *Determination of Site Earthquake Hazard*: The seismic analyses in this report made use of two data sources on the frequency of earthquakes of various intensities at the specific plant site (the seismic "hazard curve" for that site): the "Eastern United States Seismic Hazard Characterization Program," funded by the NRC at Lawrence Livermore National Laboratory (LLNL) (Ref. 2.24); and the "Seismic Hazard Methodology for the Central and Eastern United States Program," sponsored by the Electric Power Research Institute (EPRI) (Ref. 2.25). In both the LLNL and EPRI programs, seismic hazard curves were developed for all U.S. commercial power plant sites east of the Rocky Mountains using expert panels to interpret available data. The NRC staff presently considers both program results to be equally valid (Ref. 2.26). For this reason, two sets of seismic results are provided in this report. Section C.11 of Appendix C discusses the analysis of seismic hazards in more detail.

- *Identification of Accident Sequences*: The scope of the seismic analysis included loss-of-coolant accidents (LOCAs) (i.e., pipe ruptures of a spectrum of sizes including vessel rupture) and transient events. Two types of transient events were considered: those in which the power conversion system (PCS) was initially available and those in which the PCS failed as a direct consequence of the initiating event. The event trees developed in the internal-event analyses (described above) were also used to define seismically initiated accident sequences.

- *Determination of Failure Modes*: The internal-event fault trees (described above) were used in the seismic analysis, with some modification, to specify the failure modes of components, combinations of which resulted in plant system failures.

- *Determination of Fragilities*: Component seismic fragilities were obtained both from a generic fragility data base and from plant-specific fragilities estimated for components identified during a plant visit.

  The generic data base of fragility functions for seismically induced failures was originally developed as part of the Seismic Safety Margins Research Program (SSMRP) (Ref. 2.27). In that program, fragility functions for the generic categories were developed based on a combination of experimental data, design analysis reports, and an extensive survey of expert judgments, providing probability distributions of fragilities.

  Detailed fragility analyses were performed for all important structures at the studied plants. In addition, an analysis of liquefaction for the underlying soils was performed.

- *Determination of Seismic Responses*: Building and component seismic peak ground acceleration responses were computed using dynamic building models and time history analysis methods. Results from the SSMRP analysis of the Zion plant (Ref. 2.28) and methods studies (Ref. 2.23) formed the basis for assessing uncertainties in responses.

- *Computation of Core Damage Frequency*: Given the input from the five steps above, the frequencies of accident sequences, plant damage states, and core damage were

*The analysis of accident progression, containment loadings, and structural response; radioactive material transport; offsite consequences; and integrated risk for the Zion plant did not rely significantly on the previous PRA, but was essentially identical (in methods used) to the other four plant studies performed for this report.

calculated in a manner like that described above for the internal-event accident frequency analysis.

- *Estimation of Uncertainty*: The frequency distributions of individual parameters in the seismic analysis, as developed in the previous steps, were combined to yield frequency distributions of accident sequences, plant damage states, and total core damage. This process was performed using Monte Carlo techniques.

2. External-Event Methods: Fire Analysis

There were four principal steps in the fire accident frequency analysis methods used for this report. Briefly, these are:

- *Initial Plant Visit*: Based on the internal-event and seismic analyses, the general location of cables and components of the principal plant systems had previously been developed. A plant visit was then made to permit the analysis staff to see the physical arrangements in each of these areas. The analysis staff had a fire zone checklist to aid in the screening analysis and in the quantification step (described below).

  Another purpose of the initial plant visit was to confirm with plant personnel that the documentation being used was in fact the best available information and to obtain answers to questions that might have arisen in a review of the documentation. As part of this, a thorough review of firefighting procedures was conducted.

- *Screening of Potential Fire Locations*: It was necessary to select fire locations within the power plant under study that had the greatest potential for producing accident sequences of high frequency or risk. The selection of fire locations was performed using a screening analysis, which identified potentially important fire zones and prioritized these zones based on the frequencies of fire-induced initiating events in the zone and the probabilities of subsequent failures of important equipment.

- *Accident Sequence Quantification*: After the screening analysis had eliminated all but the probabilistically significant fire zones, detailed quantification of dominant accident sequences was completed as follows:

  - Determination of the temperature response in each fire zone;
  - Computation of component fire fragilities;
  - Assessment of the probability of barrier failure for the remaining combinations of fire zones; and
  - Performance of operator recovery analyses (like that described above for internal-event analyses).

- *Uncertainty Analysis*: This quantification was performed using Monte Carlo techniques like those discussed above for the internal-event analysis. No expert panels were directly used to support the development of probability distributions. Distributions for needed data were developed by the analysis staff using operating experience and experimental results.

3. External-Event Methods: Other Initiating Events

In addition to the seismic and fire external-event analyses, bounding analyses were performed for other external events that were judged to potentially contribute to the estimated plant risk. Those events that were considered included extreme winds and tornadoes, turbine missiles, internal and external flooding, and aircraft impacts.

Conservative probabilistic models were initially used in these bounding analyses. If the mean initiating event frequency resulting from such an analysis was estimated to be low (e.g., less than 1E-6 per year), the external event was eliminated from further consideration. Using this logic, the bounding analyses identified those external events in need of more study.

2.2.2 Products of Accident Frequency Analysis

The accident frequency analyses performed in this study can be displayed in a variety of ways. The specific products shown in this summary report are:

- The total core damage frequency from internal events and, where estimated, for external events.

  For Part II of this report (plant-specific results), tabular data and a histogram-type plot are used to represent the distribution of total core damage frequency. This histogram displays the fraction of Latin hypercube

sampling (LHS) observations falling within each interval.* Figure 2.2 displays an example histogram (on the right side of the figure). Four measures of the probability distribution are identified in Figure 2.2 (and throughout this report):

- Mean (arithmetic average or expected value);
- Median (50th percentile value);
- 5th percentile value; and
- 95th percentile value.

In some circumstances, the calculated probability distributions extend to very small values. When this occurs, the staff has chosen to group together all observations below a specific value. This grouped set of observations is displayed apart from (but on the same figure as) the probability distribution.

A second display of accident frequency results is used in Part III of this report, where results for all five plants are displayed together. This rectangular display (shown on the left side of Fig. 2.2) provides a summary of these four specific measures in a simple graphical form.

For those plants in which both internal and external events have been analyzed (Surry and Peach Bottom), the core damage frequency results are provided separately for internal, seismic, and fire accident initiators.

The NRC-sponsored review of the second draft of this report includes some cautions on the interpretation of low accident frequencies (Ref. 2.29). These cautions are noted on appropriate figures throughout the remainder of this report.

- The definitions and estimated frequencies of plant damage states.

  The total core damage frequency estimates described above are the sum of the frequencies of various types of accidents. For this summary report, the total core damage frequency has been divided into the contributions of plant damage states such as:**
  - Loss of all ac electric power (station blackout);
  - Transient events with failure of the reactor protection system (ATWS events);
  - Other transient events;
  - LOCAs resulting from reactor coolant system pipe ruptures, reactor coolant pump seal failures, and failed relief valves occurring within the containment building; and
  - LOCAs that bypass the containment building (steam generator tube ruptures and interfacing-system LOCAs).

Figure 2.3 is an example display of these results. In this figure, a pie chart is used to display the mean value of the total core damage frequency distribution for each of these plant damage states.

In addition to these quantitative displays, the results of the accident frequency analyses also can be discussed with respect to the qualitative perspectives obtained. In this summary report, qualitative perspectives are provided in two levels:

- *Important Plant Characteristics*: The discussion of important plant characteristics focuses on general system design and operational aspects of the plant. Perspectives are thus provided on, for example, the design and operation of the emergency diesel generators, or the capability for the "feed and bleed" mode of emergency core cooling. These results are provided in Section 3.2.2 of Chapter 3 and like numbered sections in Chapters 4 through 7.
- *Measures of Importance of Individual Events*: One typical product of a PRA is a set of "importance measures." Such measures are used to assess the relative importance of individual items (such as the failure rates of

*Care should be taken in using these histograms to estimate probability density functions. These histogram plots were developed such that the heights of the individual rectangles were not adjusted so that the rectangular areas represented probabilities. The shape of a corresponding density function may be very different from that of the histogram. The histograms represent the probability distribution of the logarithm of the core damage frequency.

**Plant damage states were defined in these risk analyses at two levels. "Summary" plant damage states were defined for use in this report and were created by combining much more detailed damage states that consider more specific types of failures and convey much more detailed information to the accident progression analysis. These more detailed plant damage states were used in the actual risk calculations. An example of the level of detail may be found in Appendix B; the contractor reports underlying this report provide and discuss the complete set of plant damage states for all plants (Refs. 2.3 through 2.7 and 2.10 through 2.14).

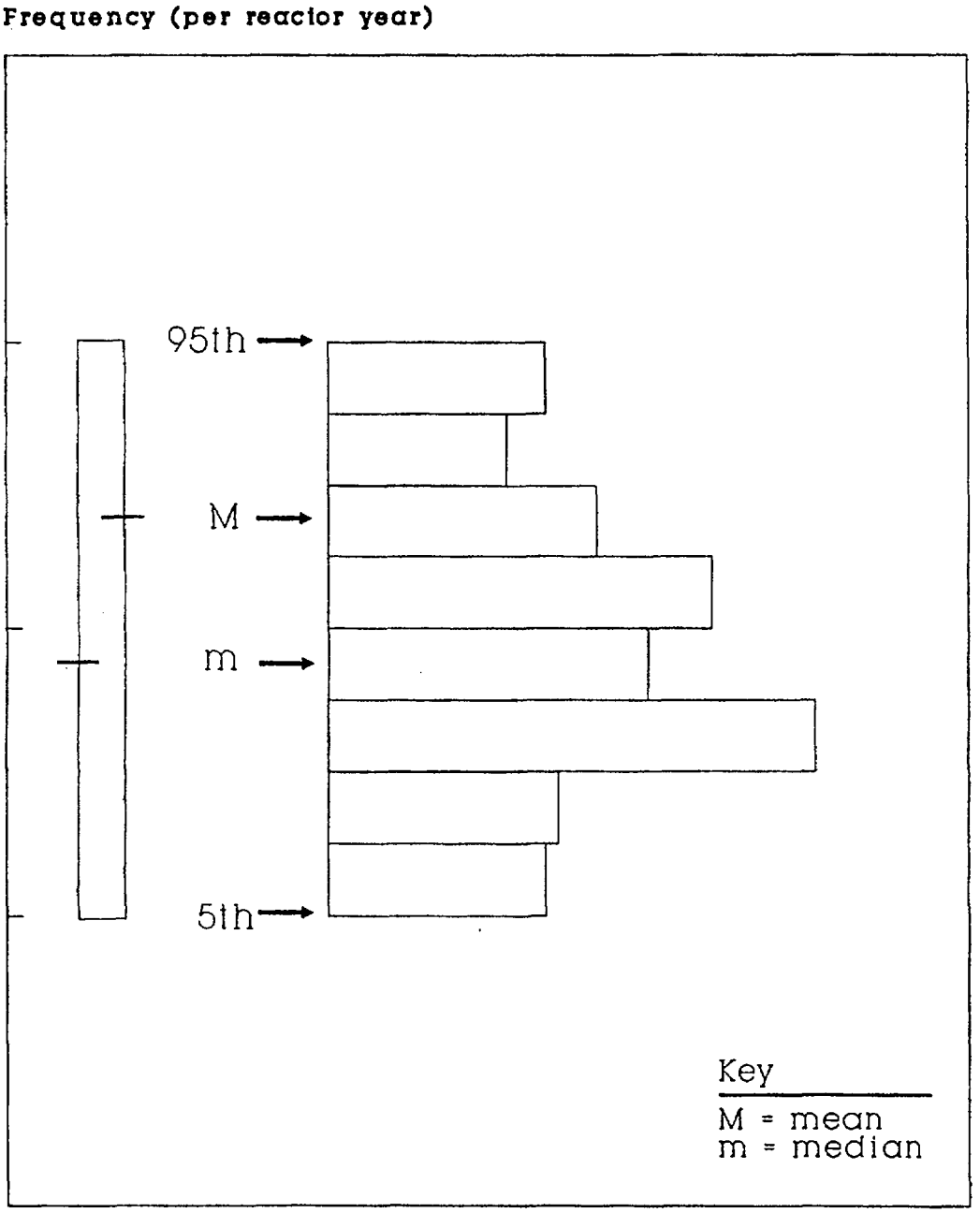


Figure 2.2 Example display of core damage frequency distribution.

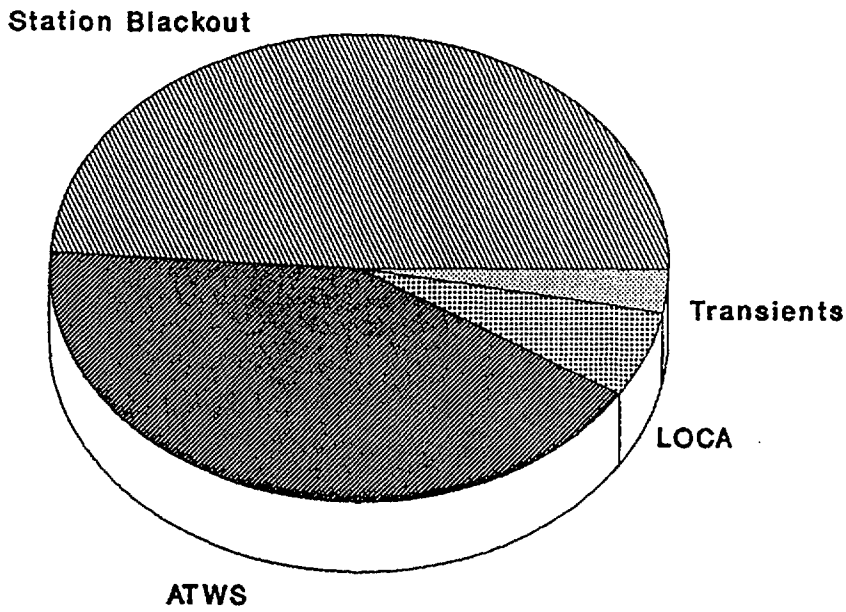


Figure 2.3 Example display of mean plant damage state frequencies.

individual plant components or the uncertainties in such failure rates) to the total core damage frequency. While a variety of measures exist, two are discussed (qualitatively) in this summary report. The first measure shows the effect of significant reductions in the frequencies of individual plant component failures or plant events (e.g., loss of offsite power, specific human errors) on the total core damage frequency. In effect, this measure shows how to most effectively reduce core damage frequency by reducing the frequencies of these individual events. The second importance measure discussed in this summary report indicates the relative contribution of key uncertainty distributions to the uncertainty in total core damage frequency. In effect, this measure shows how most effectively to reduce the uncertainty in core damage frequency by reductions in the uncertainty in individual events. These results are provided in Section 3.2.4 of Chapter 3 and like numbered sections in Chapters 4 through 7.

2.3 Accident Progression, Containment Loading, and Structural Response Analysis

2.3.1 Methods

The second part of the risk analysis process shown in Figure 2.1 ("Accident Progression, Containment Loading, and Structural Response") is the analysis of the progression of the accident after the core has begun to degrade. For each general type of accident, defined by the plant damage states, the analysis considers the important characteristics of the core melting process, the challenges to the containment building, and the response of the building to those challenges. Event trees were used to organize and quantify the large amounts of information used in this analysis. The event trees combined information from many sources, e.g., detailed computer accident simulations and panels of experts providing interpretations of available data.

In summary, the principal steps of the accident progression analysis are:

- *Development of Accident Progression Event Trees*: Accident progression event trees were used in this study to identify, sequentially order, and probabilistically quantify the important events in the progression of a severe accident. The development of an accident progression event tree consisted of identifying potentially important parameters to the accident progression and associated containment building structural response, determining possible values of each parameter (including dependencies on outcomes of previous parameters in the event tree), ordering the events chronologically, and defining the information needed to determine each parameter. The information base used consisted of accident and experimental data and calculational results from accident simulation computer codes, analyses of containment building structures, etc.* While the event tree development process used for this study is conceptually similar to that of other PRAs, both the complexity of the tree (the number of parameters and possible outcomes) and the supporting data base developed were substantially greater than those of other recent PRAs, so that more explicit use could be made of severe accident experimental and calculational information (additional discussion of the supporting data base is provided below).

- *Probabilistic Quantification of Event Trees*: Using the event tree structure and information base developed in the previous step, probability distributions for the most uncertain parameters in the accident progression event tree were generated in this step. As is typical of any PRA, this assignment of values was subjective, based on the interpretation of the data base by the risk analyst. For instance, the applicable data base is sometimes conflicting. The choice of which data to emphasize and use is a matter of each analyst's judgment, based on personal experience and familiarity. However, for this study, both the degree to which experts in accident analysis were used and the degree of documentation of the rationale for the probability distributions used were significantly greater than in other recent PRAs (additional discussion of the supporting data base is provided below).

- *Grouping of Event Tree Outcomes*: Accident progression event trees such as those constructed for this study produce a large set of alternative outcomes of a severe accident. As is typically done in PRAs, these outcomes were grouped into a smaller set of "accident progression bins." For this summary report, bins were defined principally according to the timing of containment building failure. This summary set of accident progression bins is subdivided into bins of greater detail in the supporting contractor reports (Refs. 2.10 through 2.14).

As noted above, the accident progression event trees developed for this study made extensive use of the available severe accident experimental and calculational data bases. The analysis staff made use of calculational results from a number of accident simulation computer codes, including the Source Term Code Package (Ref. 2.30), CONTAIN (Ref. 2.31), MELCOR (Ref. 2.32), and MELPROG (Ref. 2.33).

To support the analysis of certain key issues in the accident progression analysis, expert panels were convened. Fourteen accident progression, containment loadings, and structural response issues were considered by four panels, as shown in Table 2.3. These panels considered a wide range of information available from experiments and computer calculations. Using expert elicitation methods summarized in Section 2.7, probability distributions were developed based on the experts' interpretations of these issues. In addition to this set of key issues, probability distributions were developed for many other issues. Section C.1 of Appendix C provides a listing of such issues, using the Surry plant as an example. Similar listings for the other plants may be found in References 2.11 through 2.14.

Additional discussion of the methods used to develop and quantify the accident progression event trees may be found in Section A.3 of Appendix A. Reference 2.8 provides an extensive discussion of the methods used, suitable for the reader expert in severe accident and risk analysis.

Section B.3 of Appendix B provides a detailed example calculation showing how the accident progression analysis methods summarized above were used in the risk analyses supporting this report.

*In the accident progression analysis of seismic-initiated accidents, some additional loads on containment structures are considered for high-intensity earthquakes (e.g., structural loads resulting from motion of piping).

Table 2.3 Accident progression and containment structural issues evaluated by expert panels.

- In-Vessel Accident Progression Panel

  Probability of temperature-induced reactor coolant system hot leg failure (PWRs)
  Probability of temperature-induced steam generator tube failure (PWRs)
  Magnitude of in-vessel hydrogen generation (PWRs and BWRs)
  Mode of temperature-induced reactor vessel bottom head failure (PWRs and BWRs)

- Containment Loadings Panel

  Containment pressure increase at reactor vessel breach (PWRs and BWRs)
  Probability and pressure of hydrogen combustion before reactor vessel breach (Sequoyah and Grand Gulf)
  Probability and effects of hydrogen combustion in reactor building (Peach Bottom)

- Molten Core-Containment Interactions Panel

  Drywell shell meltthrough (Peach Bottom)
  Pedestal erosion from core-concrete interaction (Grand Gulf)

- Containment Structural Performance Panel

  Static containment failure pressure and mode (PWRs and BWRs)
  Probability of ice condenser failure due to hydrogen detonation (Sequoyah)
  Strength of reactor building (Peach Bottom)
  Probability of drywell and containment failure due to hydrogen detonation (Grand Gulf)
  Pedestal strength during concrete erosion (Grand Gulf)

2.3.2 Products of Accident Progression, Containment Loading, and Structural Response Analysis

The product of the accident progression and containment loading analysis is a set of accident progression bins. Each bin consists of a group of postulated accidents (with associated probabilities for each plant damage state) that has similar outcomes with respect to the subsequent portion of the risk analysis, analysis of radioactive material transport. As such, the accident progression bins are analogous to the plant damage states described in Section 2.2.1, in that they are defined based on their impact on the next analysis part. Quantitatively, the product consists of a matrix of conditional probabilities (as shown in Fig.2.4*), with the rows and columns defined by the sets of plant damage states and accident progression bins, respectively. The matrix defines the probabilities that an accident will have an outcome characteristic of a given accident progression bin if the accident began as one having the characteristic of a given plant damage state.

In this summary report, products of the accident progression analysis are shown in the following ways:

- The distribution of the probability of early containment failure** for each plant damage state.

  An example display of early containment failure probability is provided in Figure 2.5.* As may be seen, the probability distribution is represented by a histogram like that discussed above for core damage frequency.

*The mean plant damage state frequencies shown in Figures 2.4 and 2.5 (and like figures in Chapters 3 through 7) may be somewhat different from those shown in tables such as Table 3.2. The data in the latter tables resulted from uncertainty analyses using a large number of variables. The frequencies shown in the figures resulted from the uncertainty analysis of only the key accident frequency issues included in the integrated task analysis.

**In this report, early containment failure includes failures occurring before or within a few minutes of reactor vessel breach for pressurized water reactors and those failures occurring before or within 2 hours of vessel breach for boiling water reactors. Containment bypass failures are categorized separately from early failures.

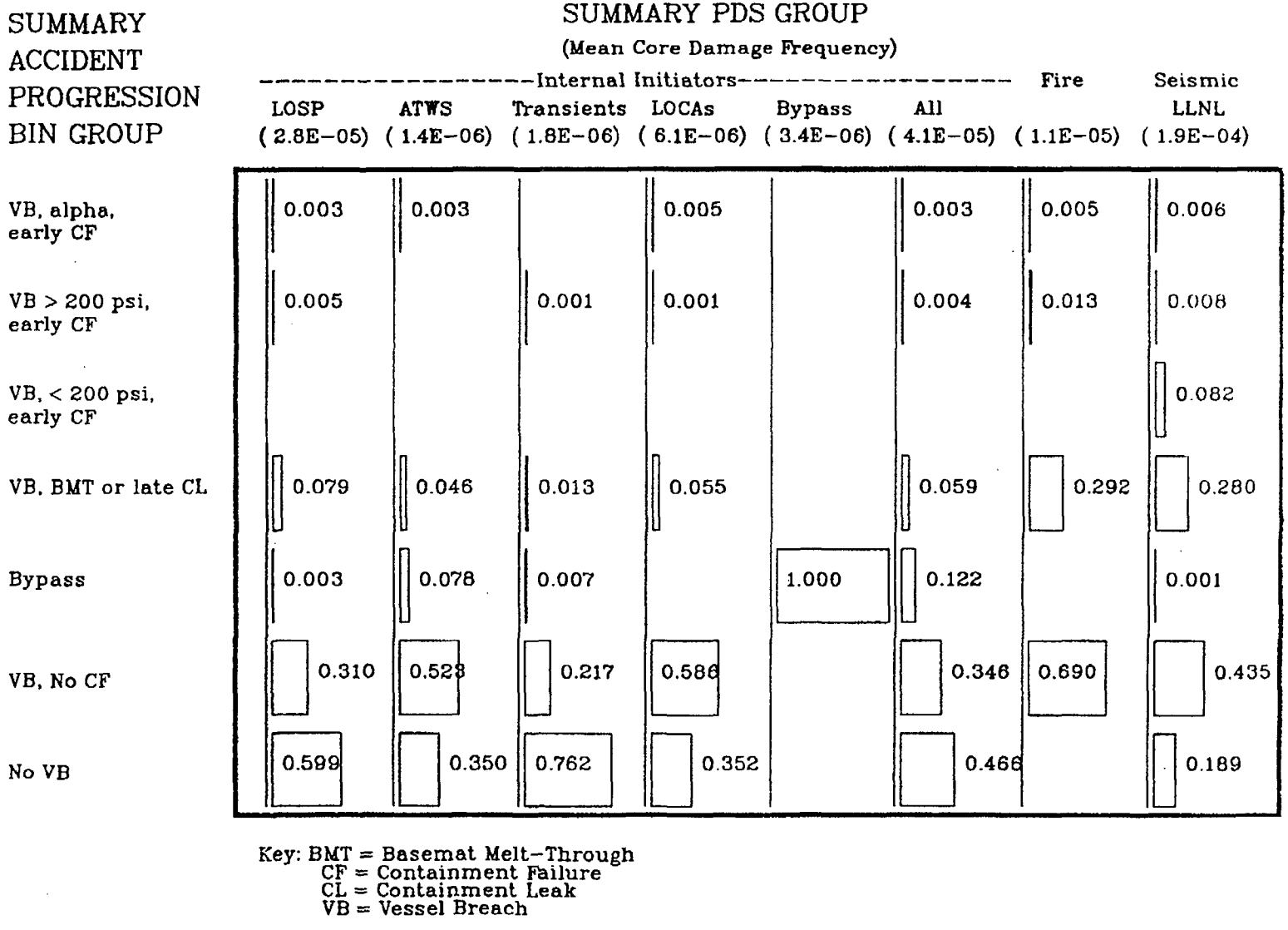


Figure 2.4 Example display of mean accident progression bin conditional probabilities.

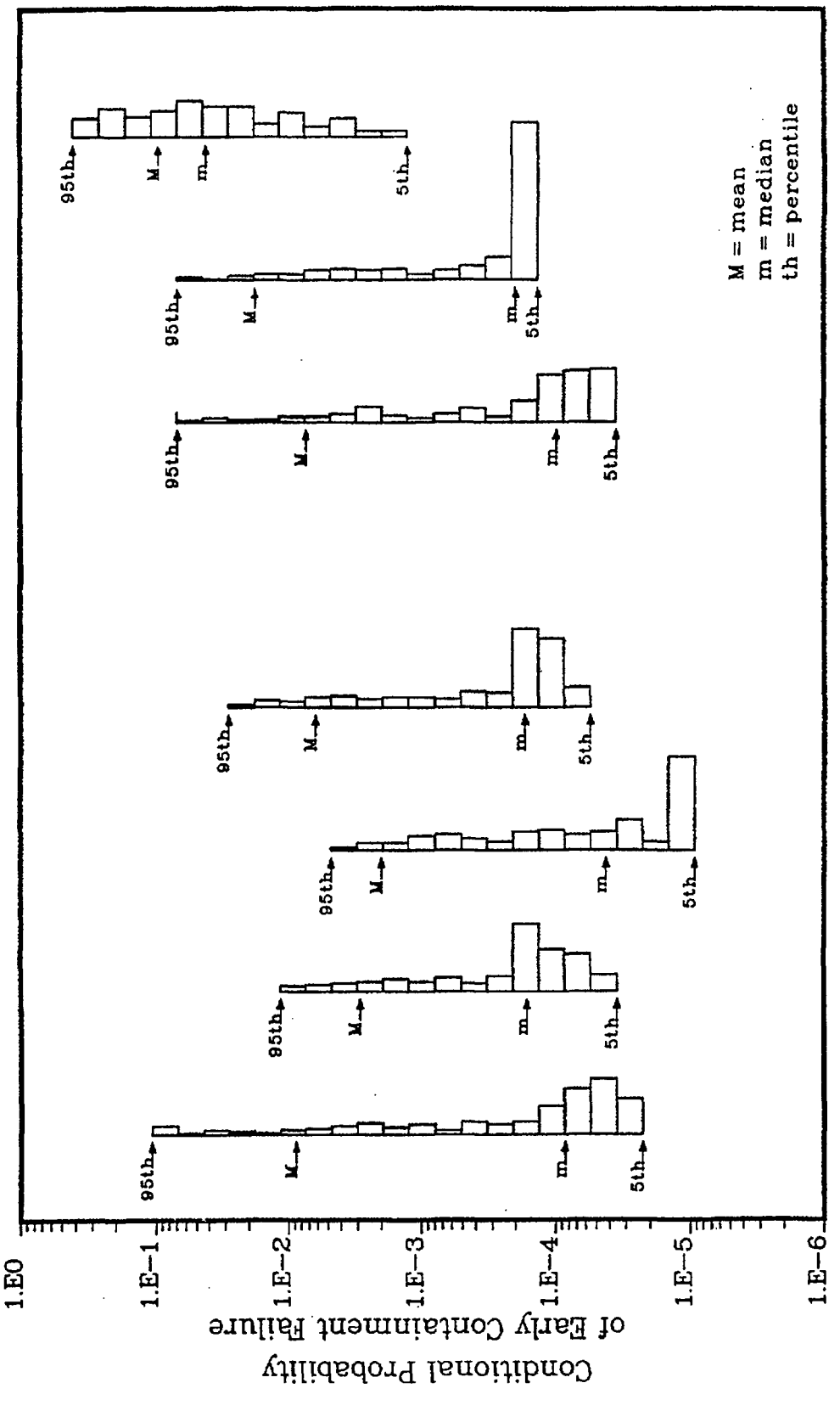


Figure 2.5 Example display of early containment failure probability distribution.

Measures of this distribution provided include:

- Mean;
- Median;
- 5th percentile value; and
- 95th percentile value.

- The mean conditional probability of each accident progression bin for each plant damage state.

  Figure 2.4 displays example results of the mean conditional probability of each accident progression bin for each plant damage state. Results are provided both in tabular and graphical (bar chart) forms.

2.4 Analysis of Radioactive Material Transport

2.4.1 Methods

The radioactive material transport analysis tracks the transport of the radioactive materials from the fuel to the reactor coolant system, then to the containment and other buildings, and finally into the environment. The fractions of the core inventory released to the atmosphere, and the timing and other release information needed to calculate the offsite consequences, together are termed the "source term." The removal and retention of radioactive material by natural processes, such as deposition on surfaces, and by engineered systems, such as sprays, are accounted for in each location.

Briefly, the principal steps in this analysis include:

- *Development of Parametric Models of Material Transport*: Because of the complexity and cost of radioactive material transport calculations performed with detailed codes, the number of accidents that could be investigated with these codes was rather limited. Further, no one detailed code available for the analyses contained models of all physical processes considered important to the risk analyses. Therefore, source terms for the variety of accidents of interest were calculated using simplified algorithms. The source terms were described as the product of release fractions and transmission factors at successive stages in the accident progression for a variety of release pathways, a variety of accident progressions, and nine classes of radionuclides. The release fraction at each stage of the accident and for each pathway is determined using various information such as predictions of detailed mechanistic codes, experimental data, etc. For the more important release parameters, listed in Table 2.4, probability distributions were developed by a panel of experts. The set of codes (one for each plant) used to calculate the source terms is known collectively as the "XSOR" codes (Ref. 2.34). The XSOR codes are parametric in nature; that is, they are designed to use the results of more detailed mechanistic codes or analyses as input.

Table 2.4 Source term issues evaluated by expert panel.

• Source Term Expert Panel
In-vessel retention and release of radioactive material (PWRs and BWRs)
Revolatization of radioactive material from the reactor vessel and reactor coolant system (early and late) (PWRs and BWRs)
Radioactive releases during high-pressure melt ejection/direct containment heating (PWRs and BWRs)
Radioactive releases during core-concrete interaction (PWRs and BWRs)
Retention and release from containment of core-concrete interaction radioactive releases (PWRs and BWRs)
Ice condenser decontamination factor (Sequoyah)
Reactor building decontamination factor (Grand Gulf)
Late sources of iodine (Grand Gulf)

Release terms are divided into two time periods, an early release and a delayed release. The timing of release is particularly important for the prediction of early health effects.

- *Detailed Analysis of Radioactive Material Transport for Selected Accident Progression Bins*: Once the basic XSOR algorithm was defined, it was necessary to insert parameters analogous to the quantification of the accident progression event tree in the previous part of the analysis. Since a quantitative uncertainty analysis was one of the objectives of this study, data on the more important parameters were constructed in the form of probability distributions. These distributions were developed based on calculations from the Source Term Code Package (STCP) (Ref. 2.30), CONTAIN (Ref. 2.31), MELCOR (Ref. 2.32), and other calculational and experimental data. The source term parameters determined by an expert panel are shown in Table 2.4. Distributions for parameters that were judged of lesser importance were evaluated by experts drawn from the analysis staff or from other groups at national laboratories. (See Section C.1 of Appendix C for a listing of such parameters for the Surry plant. Similar listings for the other plants may be found in Refs. 2.11 through 2.14.) In rare instances, single-valued estimates were used.

- *Grouping of Radioactive Releases*: For these risk analyses, radioactive releases were grouped according to their potential to cause early and latent cancer fatalities and warning time.* Through this "partitioning" process, the large number of radioactive releases calculated with the XSOR codes were collected into a small set of source term groups (30 to 60 in number). This set of groups was then used in the offsite consequence calculations discussed below.

Additional discussion of the methods used to perform the radioactive material transport analysis may be found in Section A.4 of Appendix A. Reference 2.8 provides an extensive discussion of the methods used that is suitable for the reader expert in severe accident and risk analysis.

Section B.4 of Appendix B provides a detailed example calculation showing how the radioactive material transport analysis methods summarized above were used in the risk analyses supporting this report.

*This grouping of source terms by offsite consequence effects is analogous to the grouping of accident sequences into plant damage states by their potential effect on accident progression.

2.4.2 Products of Radioactive Material Transport Analysis

The product of this part of the risk analysis is the estimate of the radioactive release magnitude, with associated energy content, time, elevation, and duration of release, for each of the specified source term groups developed in the "partitioning" process described above.

The radioactive release estimates generated in this part of the risk analysis can be displayed in a variety of ways. In this report, radioactive release magnitudes are shown in the following ways:

- Distribution of release magnitudes for each of the nine isotopic groups for selected accident progression bins.

  The results of the radioactive material transport analysis can vary in form depending on the intended use. For purposes of this report, example results that display the distribution of release magnitudes for selected accident progression bins were obtained. In Part II of this report, the results for two accident progression bins are displayed for each plant. For these selected accident progression bins, the distribution of the radioactive release magnitude (for each of the nine radionuclide groups) is characterized by the mean, median, 5th percentile, and 95th percentile. An example distribution is displayed in Figure 2.6. (Distributions of this type are constructed with the assumption that all estimated source terms are equally likely and thus do not incorporate the frequencies of the individual source terms. Recalculation of these distributions, including consideration of frequencies, does not significantly change the results.)

- Frequency distribution of radioactive releases of iodine, cesium, strontium, and lanthanum.

  Chapter 10 displays the absolute frequency* of source term release magnitudes. These results are presented in the form of complementary cumulative distribution functions (CCDFs) of the magnitude of iodine, cesium, strontium, and lanthanum releases.** This

*That is, the combined frequency of all plant damage state frequencies and conditional accident progression bin probabilities.

**These four groups are used to represent the spectrum of possible chemical groups, i.e., from chemically volatile to nonvolatile species.

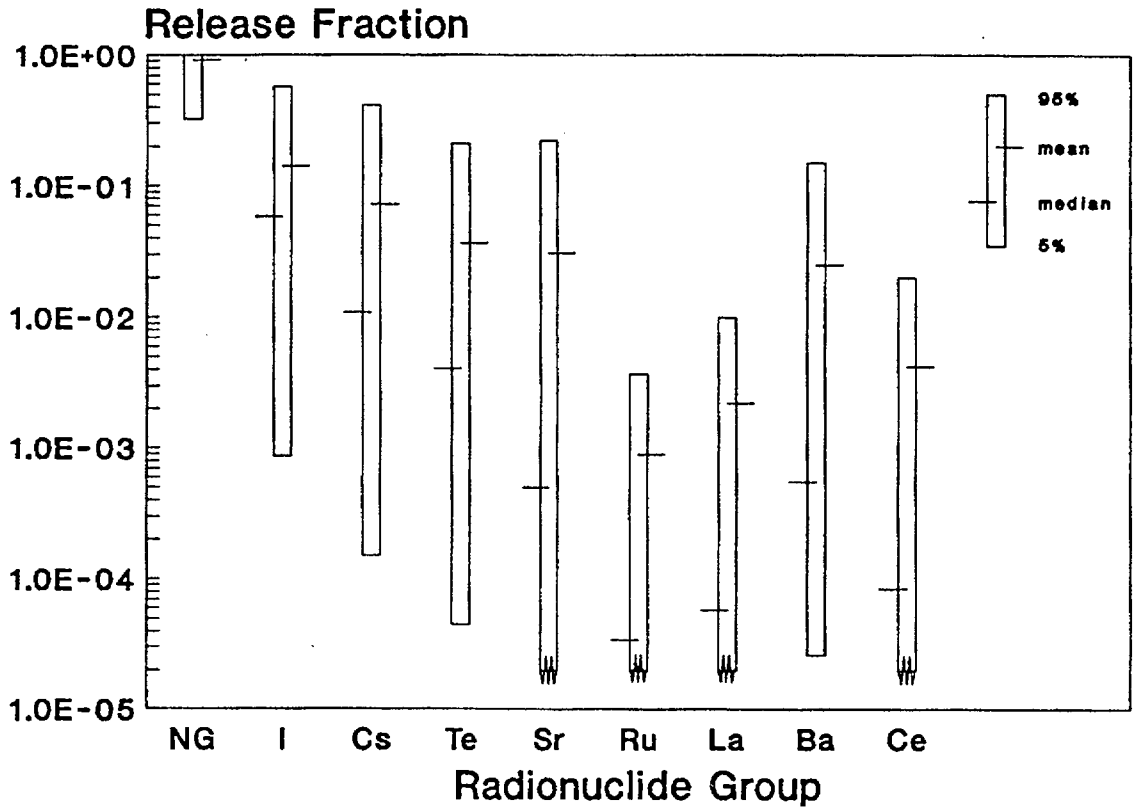


Figure 2.6 Example display of radioactive release distributions.

display provides information on the frequency of source term magnitudes exceeding a specific value for each of the plants. Figure 2.7 displays an example CCDF for one chemical group.

2.5 Offsite Consequence Analysis

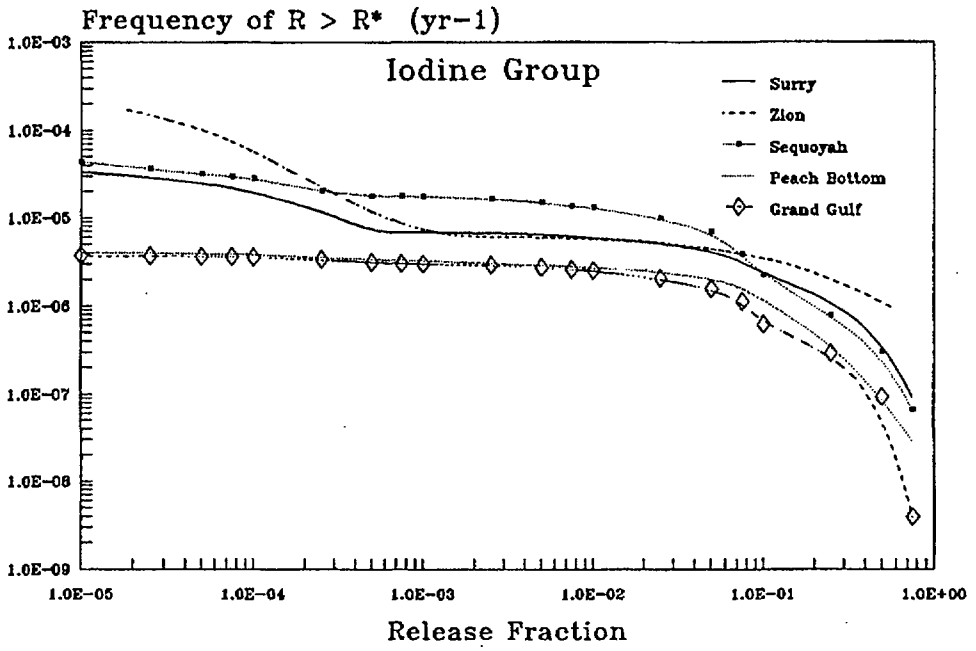
2.5.1 Methods

The severe accident radioactive releases described in the preceding section are of concern because of their potential for impacts on the surrounding environment and population. The impacts of such releases to the atmosphere can manifest themselves in a variety of early and delayed health effects, loss of habitability of areas close to the plant site, and economic losses. The fourth part of the risk analysis process shown in Figure 2.1 represents the estimation of these offsite consequences, given the radioactive releases (source term groups) generated in the previous analysis part.

There are five principal steps in the offsite consequence analysis. Briefly, these are:

- *Assessment of Pre-accident Inventories of Radioactive Material*: An assessment was made of the pre-accident inventories of each radioactive species in the reactor fuel, using information on the thermal power and refueling cycles for the plants studied. For the source term and offsite consequence analysis, the radioactive species were collected into groups of similar chemical behavior. For these risk analyses, nine groups were used to represent 60 radionuclides considered to be of most importance to offsite consequences: noble gases, iodine, cesium, tellurium, strontium, ruthenium, cerium, barium, and lanthanum.

- *Analysis of Transport and Dispersion of Radioactive Material*: The transport and dispersion of radioactive material to offsite



Note: As discussed in Reference 2.29, 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 2.7 Example display of source term complementary cumulative distribution function.

areas was modeled in two parts: the initial development of a plume in the wake of plant buildings, using models described in Reference 2.35; and the subsequent downwind transport, which used a straight-line Gaussian plume model, as described in Reference 2.36. The effect of the initial sensible energy content of the plume was included in these models so that under some conditions plume "liftoff" could occur, elevating the contained radioactive material into the atmosphere.

The dispersion models used in this report also explicitly accounted for the variability of transport and deposition with weather conditions.

Meteorological data for each specific power plant site were used. For each of a set of approximately 160 representative weather conditions, a dispersion pattern of the plume was calculated. Deposition of radioactive material from the plume onto the ground (or water bodies) beneath the plume was based on a set of experimentally derived deposition rates for dry and wet (rain) conditions.

- *Analysis of the Radiation Doses*: Using the dispersion and deposition patterns developed in the previous step and a set of dose conversion factors (which relate a concentration of a radioactive species to a dose to a given body organ) (Refs. 2.37, 2.38, and 2.39), calculations were made of the doses received by the exposed populations via direct (cloudshine, inhalation, groundshine) and indirect (ingestion, resuspension of radioactive material from the ground into the air) pathways. Site-specific population data were used in these calculations. The doses were calculated on a body organ-by-organ basis and combined into health effect estimates in a later step.

- *Analysis of Dose Mitigation by Emergency Response Actions*: Consideration was given to the mitigating effects of emergency response actions taken immediately after the accident and in the longer term. Effects included were evacuation, sheltering, and relocation of people, interdiction of milk and crops, and decontamination, temporary interdiction, and/or condemnation of land and buildings.

  The analysis of offsite consequences for this study included a "base case" and several sets of alternative emergency response actions. For the base case, it was assumed that 99.5 percent of the population within the 10-mile emergency planning zone (EPZ) participated in an evacuation. This set of people was assumed to move away from the plant site at a speed estimated from the plant licensee's emergency plan, after an initial delay (to reach the decision to evacuate and permit communication of the need to evacuate) also estimated from the licensee's plan. It was also assumed that the 0.5 percent of the population that did not participate in the initial evacuation was relocated within 12 to 24 hours after plume passage, based on the measured concentrations of radioactive material in the surrounding area and the comparison of projected doses with proposed Environmental Protection Agency (EPA) guidelines (Ref. 2.40). Similar relocation assumptions were made for the population outside the 10-mile planning zone. Longer-term countermeasures (e.g., crop or land interdiction) were based on EPA and Food and Drug Administration guidelines (Ref. 2.41).

  Several alternative emergency response assumptions were also analyzed in this study's offsite consequence and risk analyses. These included:

  - Evacuation of 100 percent of the population within the 10-mile emergency planning zone;
  - Indoor sheltering of 100 percent of the population within the EPZ (during plume passage) followed by rapid subsequent relocation after plume passage;
  - Evacuation of 100 percent of the population in the first 5 miles of the planning zone, and sheltering followed by fast relocation of the population in the second 5 miles of the EPZ; and
  - In lieu of evacuation or sheltering, only relocation from the EPZ within 12 to 24 hours after plume passage, using relocation criteria described above.

  In each of these alternatives, the region outside the 10-mile zone was subject to a common assumption that relocation was performed based on comparisons of projected doses with EPA guidelines (as discussed above).

- *Calculation of Health Effects*: The offsite consequence analysis calculated the following health effect measures:

  - The number of early fatalities and early injuries expected to occur within 1 year of the accident and the latent cancer fatalities expected to occur over the lifetime of the exposed individuals;
  - The total population dose received by the people living within specific distances (e.g., 50 miles) of the plant; and
  - Other specified measures of offsite health effect consequences (e.g., the number of early fatalities in the population living within 1 mile of the reactor site boundary).

  The health effects calculated in this analysis were based on the models of Reference 2.42. This work in turn used the work of the BEIR III report (Ref. 2.43) for its models of latent cancer effects.

The schedule for completing the risk analyses of this report did not permit the performance of uncertainty analyses for parameters of the offsite consequence analysis, although variability due to annual variations in meteorological conditions is included. Such an analysis is, however, planned to be performed.

Section A.5 of Appendix A provides additional discussion of the methods used for performing the offsite consequence analysis. The reader seeking extensive discussion of the methods used is directed to Reference 2.8 and to Reference 2.36, which discusses the computer code used to perform the offsite consequence analysis (i.e., the MELCOR Accident Consequence Code System (MACCS), Version 1.5).

2.5.2 Products of Offsite Consequence Analysis

The product of this part of the risk analysis process is a set of offsite consequence measures for

each source term group. For this report, the specific consequence measures discussed include early fatalities, latent cancer fatalities, total population dose (within 50 miles and entire site region), and two measures for comparison with NRC's safety goals (average individual early fatality probability within 1 mile and average individual latent cancer fatality probability within 10 miles of the site boundary) (Ref. 2.44).

For display in this report, the results of the offsite consequence analyses are combined with the frequencies generated in the previous analysis steps and shown in the form of complementary cumulative distribution functions (CCDFs). This display shows the frequency of consequences occurring at a level greater than a specified amount. Figure 2.8 provides a display of such a CCDF. This information is also provided in tabular form in Chapter 11.

2.6 Uncertainty Analysis

As stated in the introduction to the chapter, an important characteristic of the probabilistic risk analyses conducted in support of this report is that they have explicitly included an estimation of the uncertainties in the calculations of core damage frequency and risk that exist because of incomplete understanding of reactor systems and severe accident phenomena.

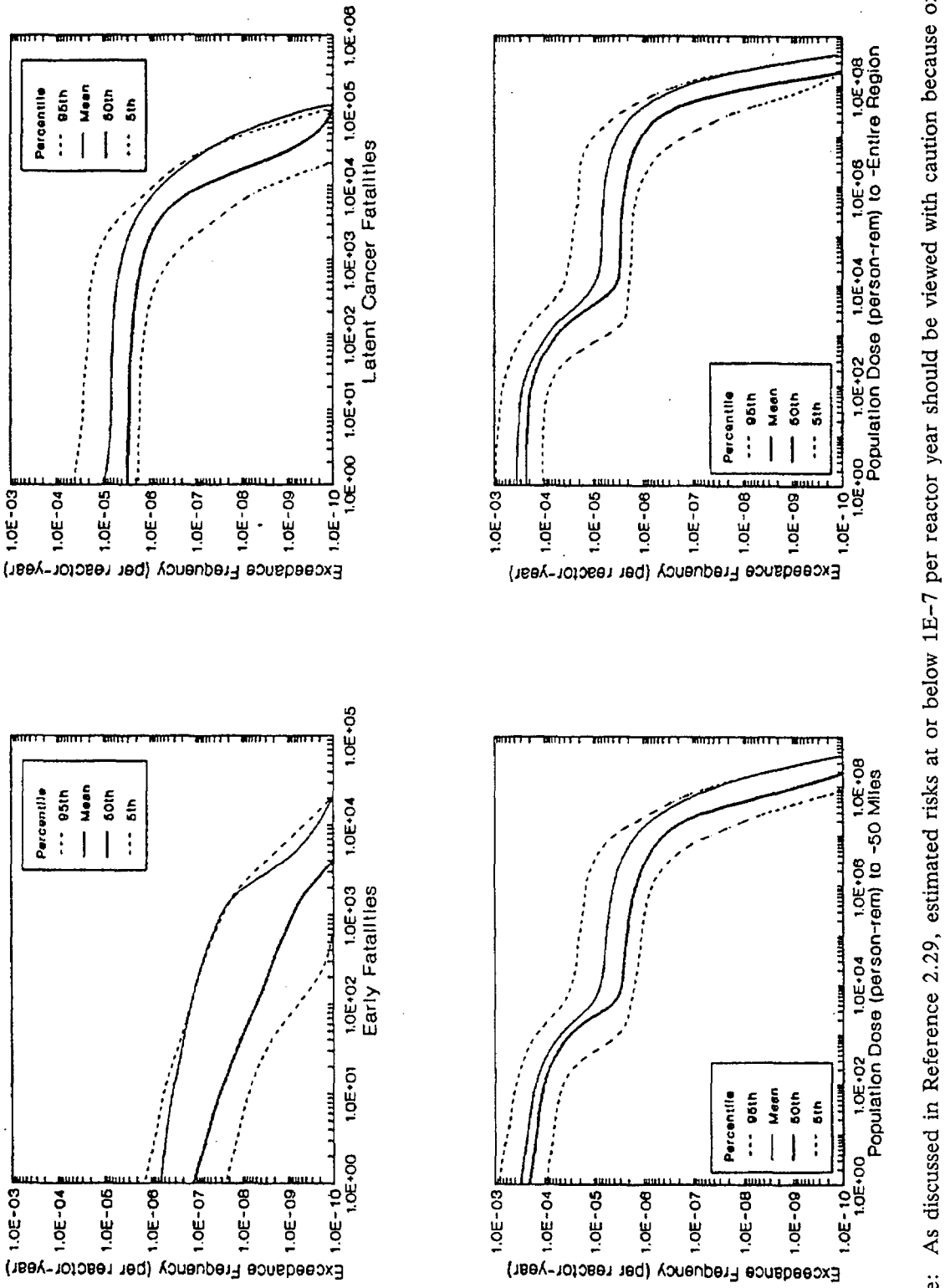
There are four steps in the performance of uncertainty analyses. Briefly, these are:

- *Scope of Uncertainty Analyses*: Important sources of uncertainty exist in all four stages of the risk analysis shown in Figure 2.1. In this study, the total number of parameters that could be varied to produce an estimate of the uncertainty in risk was large, and it was somewhat limited by the computer capacity required to execute the uncertainty analyses. Therefore, only the most important sources of uncertainty were included. Some understanding of which uncertainties would be most important to risk was obtained from previous PRAs, discussion with phenomenologists, and limited sensitivity analyses. Subjective probability distributions for parameters for which the uncertainties were estimated to be large and important to risk and for which there were no widely accepted data or analyses were generated by expert panels. Those issues for which expert panels generated probability distributions are listed in Tables 2.2 through 2.4.

- *Definition of Specific Uncertainties*: In order for uncertainties in accident phenomena to be included in the probabilistic risk analyses conducted for this study, they had to be expressed in terms of uncertainties in the parameters that were used in the study. Each section of the risk analysis was conducted at a slightly different level of detail. However, each analysis part (except for offsite consequence analysis, which was not included in the uncertainty analysis) did not calculate the characteristics of the accidents in as much detail as would a mechanistic and detailed computer code. Thus, the uncertain input parameters used in this study are "high level" or summary parameters. The relationships between fundamental physical parameters and the summary parameters of the risk analysis parts are not always clear; this lack of understanding leads to what is referred to in this study as modeling uncertainties. In addition, the values of some important physical or chemical parameters are not known and lead to uncertainties in the summary parameters. These uncertainties were referred to as data uncertainties. Both types of uncertainties were included in the study, and no consistent effort was made to differentiate between the effects of the two types of uncertainties.

  Parameters were chosen to be included in the uncertainty analysis if the associated uncertainties were estimated to be large and important to risk.

- *Development of Probability Distributions*: Probability distributions for input parameters were developed by a number of methods. As stated previously, distributions for many key input parameters were determined by panels of experts. The experts used a large variety of techniques to generate probability distributions, including reliance on detailed code calculations, extrapolation of existing experimental and accident data to postulated conditions during the accident, and complex logic networks. Probability distributions were obtained from the expert panels using formalized procedures designed to minimize bias and maximize accuracy and scrutability of the experts' results. These procedures are described in more detail in Section 2.7. Probability distributions for some parameters believed to be of less importance to risk were generated by analysts on the project staff or by phenomenologists from several different



Note: As discussed in Reference 2.29, 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 2.8 Example display of offsite consequences complementary cumulative distribution function.

national laboratories using techniques like those employed with the expert panels. (Section C.1 of Appendix C provides a listing of parameters to which probability distributions were assigned for the Surry plant. Similar listings for the other plants may be found in Refs. 2.11 through 2.14.)

Probability distributions for many of the most important accident sequence frequency variables were generated using statistical analyses of plant data or data from other published sources.

- *Combination of Uncertainties*: A specialized Monte Carlo method, Latin hypercube sampling, was used to sample the probability distributions defined for the many input parameters. The sample observations were propagated through the constituent analyses to produce probability distributions for core damage frequency and risk. Monte Carlo methods produce results that can be analyzed with a variety of techniques, such as regression analysis. Such methods easily treat distributions with wide ranges and can incorporate correlations between variables. Latin hypercube sampling (Ref. 2.20) provides for a more efficient sampling technique than straightforward Monte Carlo sampling while retaining the benefits of Monte Carlo techniques. It has been shown to be an effective technique when compared to other, more costly, methods (Ref. 2.45). Since many of the probability distributions used in the risk analyses are subjective distributions, the composite probability distributions for core damage frequency and risk must also be considered subjective.

Additional discussion of uncertainty analysis methods is provided in Section A.6 of Appendix A and in detail in Reference 2.8.

2.7 Formal Procedures for Elicitation of Expert Judgment

The risk analysis of severe reactor accidents inherently involves the consideration of parameters for which little or no experiential data exist. Expert judgment was needed to supplement and interpret the available data on these issues. The elicitation of experts on key issues was performed using a formal set of procedures, discussed in greater detail in Reference 2.8. The principal steps of this process are shown in Figure 2.9. Briefly, these steps are:

- *Selection of Issues*: As stated in Section 2.6, the total number of uncertain parameters that could be included in the core damage frequency and risk uncertainty analyses was somewhat limited. The parameters considered were restricted to those with the largest uncertainties, expected to be the most important to risk, and for which widely accepted data were not available. In addition, the number of parameters that could be determined by expert panels was further restricted by time and resource limitations. The parameters that were determined by expert panels are, in the vernacular of this project, referred to as "issues." An initial list of issues was chosen from the important uncertain parameters by the plant analyst, based on results from the first draft NUREG-1150 analyses (Ref. 2.46). The list was further modified by the expert panels. Tables 2.2 through 2.4 list those issues studied by expert panels.

- *Selection of Experts*: Seven panels of experts were assembled to consider the principal issues in the accident frequency analyses (two panels), accident progression and containment loading analyses (three panels), containment structural response analyses (one panel), and source term analyses (one panel). The experts were selected on the basis of their recognized expertise in the issue areas, such as demonstrated by their publications in refereed journals. Representatives from the nuclear industry, the NRC and its contractors, and academia were assigned to panels to ensure a balance of "perspectives." Diversity of perspectives has been viewed by some (e.g., Refs. 2.47 and 2.48) as allowing the problem to be considered from more viewpoints and thus leading to better quality answers. The size of the panels ranged from 3 to 10 experts.

- *Training in Elicitation Methods*: Both the experts and analysis team members received training from specialists in decision analysis. The team members were trained in elicitation methods so that they would be proficient and consistent in their elicitations. The experts' training included an introduction to the elicitation and analysis methods, to the psychological aspects of probability estimation (e.g., the tendency to be overly confident in the estimation of probabilities), and to probability estimation. The purpose of this training was to better enable the experts to transform their knowledge and judgments into the form of probability distributions and to avoid

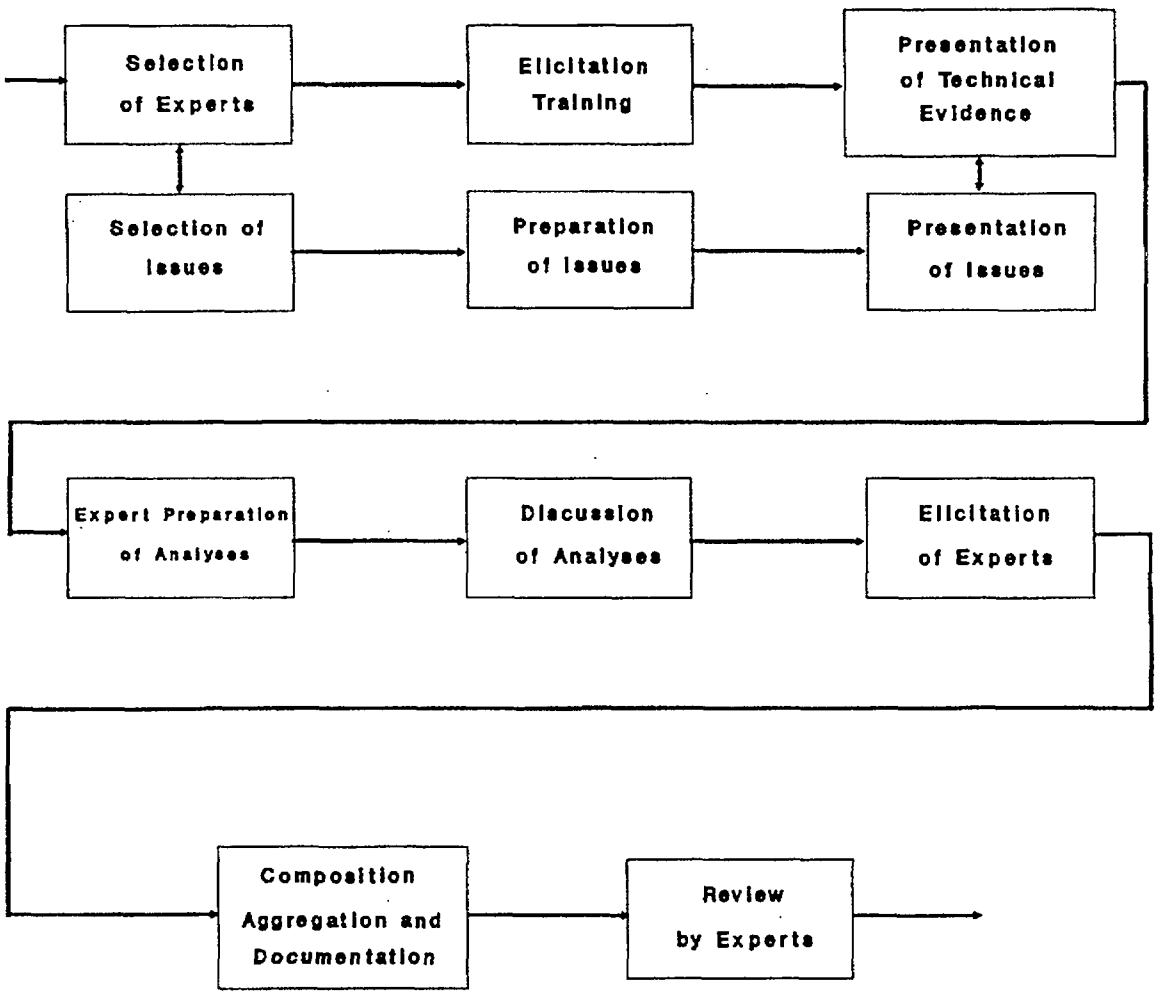


Figure 2.9 Principal steps in expert elicitation process.

particular psychological biases such as overconfidence. Additionally, the experts were given practice in assigning probabilities to sample questions with known answers (almanac questions). Studies such as those discussed in Reference 2.49 have shown that feedback on outcomes can reduce some of the biases affecting judgmental accuracy.

- *Presentation and Review of Issues*: Presentations were made to each panel on the set of issues to be considered, the definition of each issue, and relevant data on each issue. Other parameters considered by the analysis staff to be of somewhat lesser importance were also described to the experts. The purposes of these presentations were to permit the panel to add or drop issues depending on their judgments as to their importance; to provide a specific definition of each issue chosen and the sets of associated boundary conditions imposed by other issue definitions; and to obtain information from additional data sources known to the experts.

  In addition, written descriptions of the issues were provided to the experts by the analysis staff. The descriptions provided the same information as provided in the presentations, in addition to reference lists of relevant technical material, relevant plant data, detailed descriptions of the types of accidents of most importance, and the context of the issue within the total analysis. The written descriptions also included suggestions of how the issues could be decomposed into their parts using logic trees. The issues were to be decomposed because the decomposition of problems has been shown to ease the cognitive burden of considering complex problems and to improve the accuracy of judgments (Ref. 2.50).

  For the initial meeting, researchers, plant representatives, and interested parties were invited to present their perspectives on the issues to the experts. Frequently, these presentations took several days.

- *Preparation of Expert Analyses*: After the initial meeting at which the issues were presented, the experts were given time to prepare their analyses of the issues. This time ranged from 1 to 4 months. The experts were encouraged to use this time to investigate alternative methods for decomposing the issues, to search for additional sources of information on the issues, and to conduct calculations. During this period, several panels met to exchange information and ideas concerning the issues. During some of these meetings, expert panels were briefed by the project staff on the results from other expert panels in order to provide the most current data.

- *Expert Review and Discussion*: After the expert panels had prepared their analyses, a final meeting was held in which each expert discussed the methods he/she used to analyze the issue. These discussions frequently led to modifications of the preliminary judgments of individual experts. However, the experts' actual judgments were not discussed in the meeting because group dynamics can cause people to unconsciously alter their judgments in the desire to conform (Ref. 2.51).

- *Elicitation of Experts*: Following the panel discussions, each expert's judgments were elicited. These elicitations were performed privately, typically with an individual expert, an analysis staff member trained in elicitation techniques, and an analysis staff member familiar with the technical subject. With few exceptions, the elicitations were done with one expert at a time so that they could be performed in depth and so that an expert's judgments would not be adversely influenced by other experts. Initial documentation of the expert's judgments and supporting reasoning were obtained in these sessions.

- *Composition and Aggregation of Judgments*: Following the elicitation, the analysis staff composed probability distributions for each expert's judgments. The individual judgments were then aggregated to provide a single composite judgment for each issue. Each expert was weighted equally in the aggregation because this simple method has been found in many studies (e.g., Ref. 2.52) to perform the best.

- *Review by Experts*: Each expert's probability distribution and associated documentation developed by the analysis staff was reviewed by that expert. This review ensured that potential misunderstandings were identified and corrected and that the issue documentation properly reflected the judgments of the expert.

2.8 Risk Integration

2.8.1 Methods

The fifth part of the risk analysis process shown in Figure 2.1 ("Risk Integration") is the integration of the other analysis products into the overall estimate of plant risk. Risk for a given consequence measure is the sum over all postulated accidents of the product of the frequency and consequence of the accident. This part of the analysis consisted of both the combination of the results of the constituent analyses and the subsequent assessment of the relative contributions of different types of accidents (as defined by the plant damage states, accident progression bins, or source term groups) to the total risk.

Appendix A provides a more detailed description of the risk integration process. In order to assist the reader seeking a detailed understanding of this process, an example calculation is provided in Appendix B. This example makes use of actual results for the Surry plant.

2.8.2 Products of Risk Integration

The risk analyses performed in this study can be displayed in a variety of ways. The specific products shown in this summary report are described below, with similar products provided for early fatality risk, latent cancer fatality risk, population dose risk within 50 miles and within the entire area surrounding the site, and for two measures related to NRC's safety goals (Ref. 2.44).

- The total risks from internal and fire events.*

  Reflecting the uncertain nature of risk results, such results can be displayed using a probability density function. For Part II of this report (plant-specific results), a histogram is used. This histogram for risk results is like that shown on the right side of Figure 2.2 for the results of the accident frequency analysis. In addition, four measures of the probability distribution are identified in Figure 2.2 (and throughout this report):

  - Mean;
  - Median;
  - 5th percentile value; and
  - 95th percentile value.

  A second display of risk results is used in Part III of this report, where results for all five plants are displayed together. This rectangular display (shown on the left side of Fig. 2.2) provides a summary of these four specific measures in a simple graphical form.

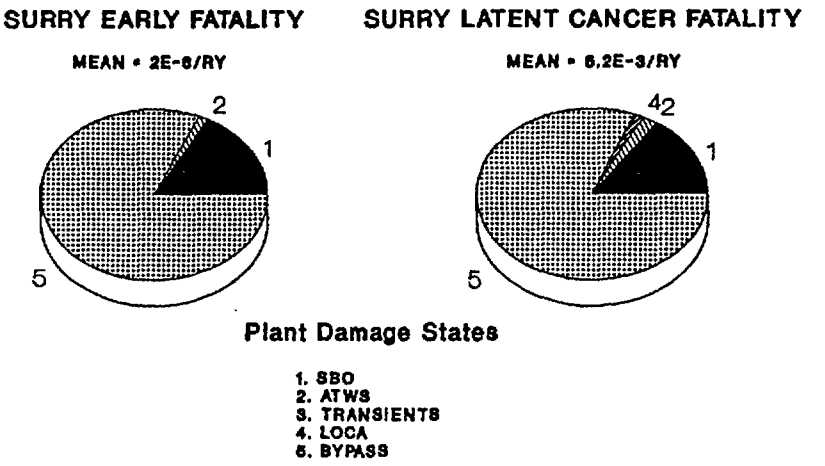
- Contributions of plant damage states and accident progression bins to mean risk.

  The risk results generated in this report can be decomposed to determine the fractional contribution of individual plant damage states and accident progression bins to the mean risk. An example display of the fractional contribution of plant damage states to mean early and latent cancer fatality risk is provided in Figure 2.10. The estimated values of these relative contributions are somewhat sensitive to the Monte Carlo sampling variation, particularly those contributions that are small. References 2.10 through 2.14 discuss this sensitivity to sampling variation in more detail. These references also include discussion of an alternative method for calculating the relative contributions to mean risk that provides somewhat different results.

- Contributions to risk uncertainty.

  Regression analyses were performed to assess the relative contributions of the uncertainty in individual parameters (or groups of parameters) to the uncertainty in risk. Results of these analyses are discussed in Part III of this report and in more detail in References 2.10 through 2.14.

*For reasons described in Chapter 1, seismic risk is not displayed or discussed in this report.



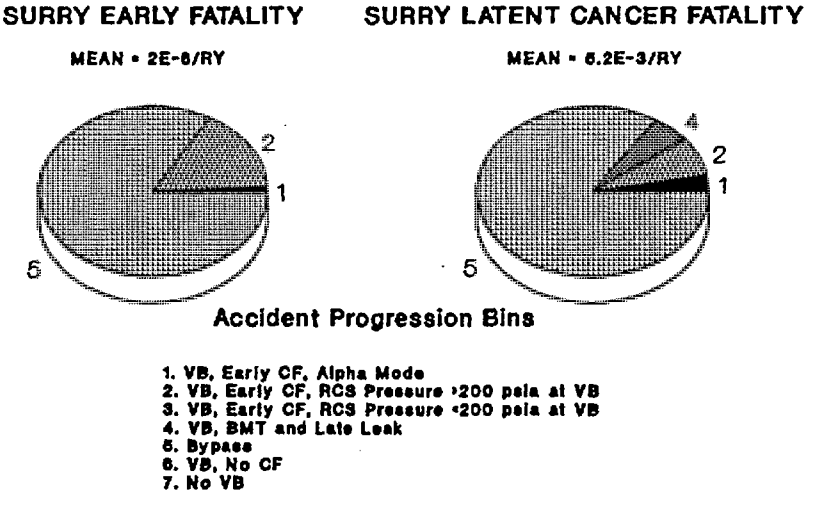


Figure 2.10 Example display of relative contributions to mean risk.

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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.

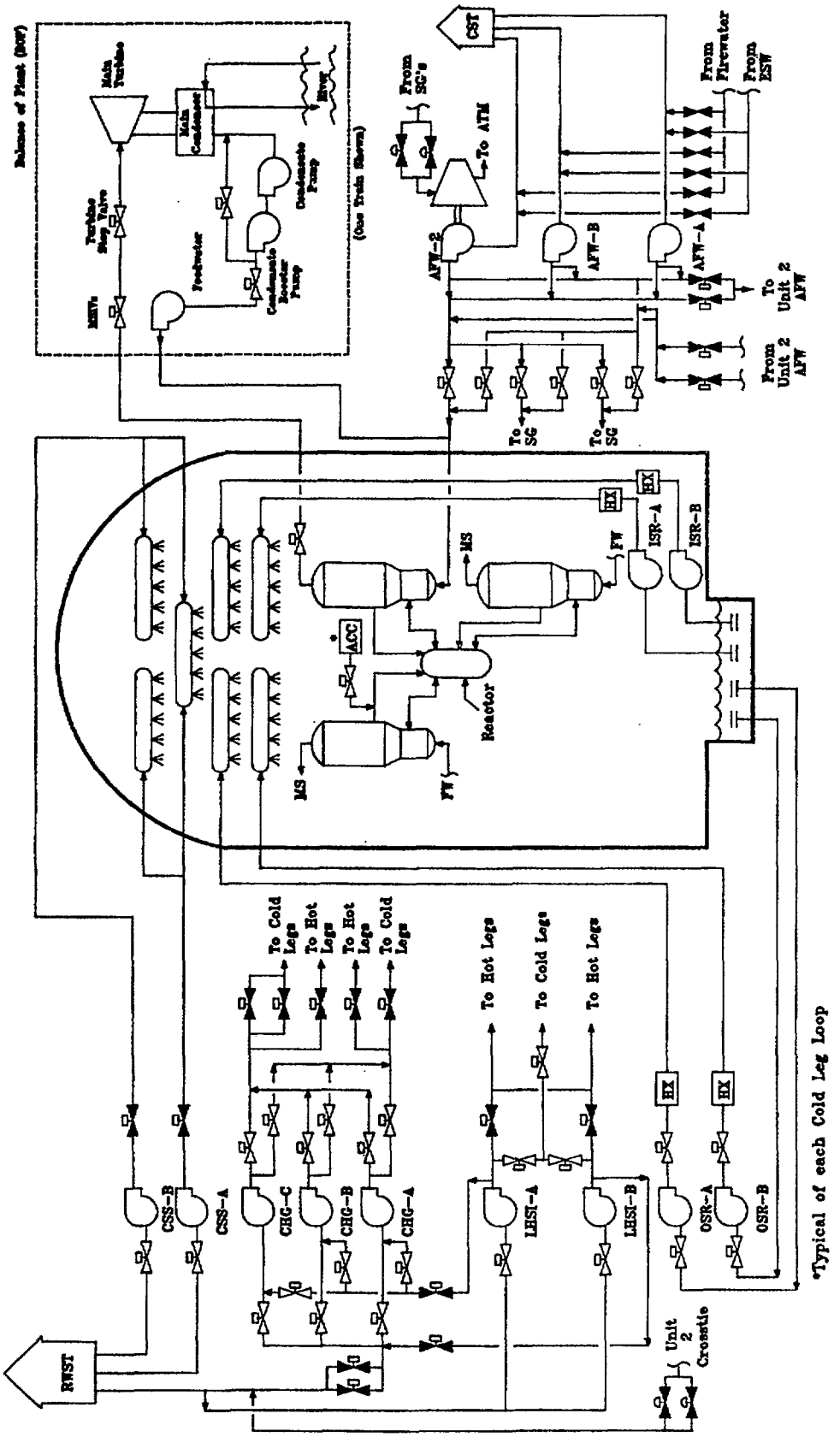


Figure 3.1 Surry plant schematic.

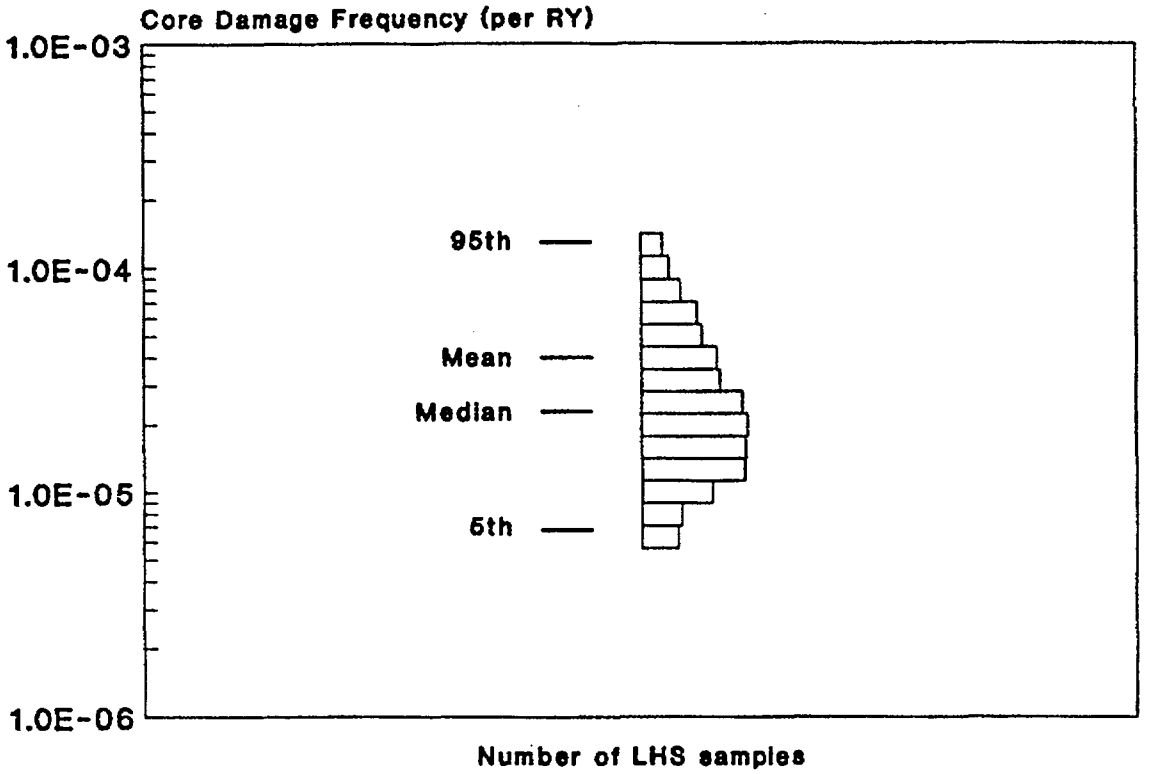


Figure 3.2 Internal core damage frequency results at Surry.*

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.

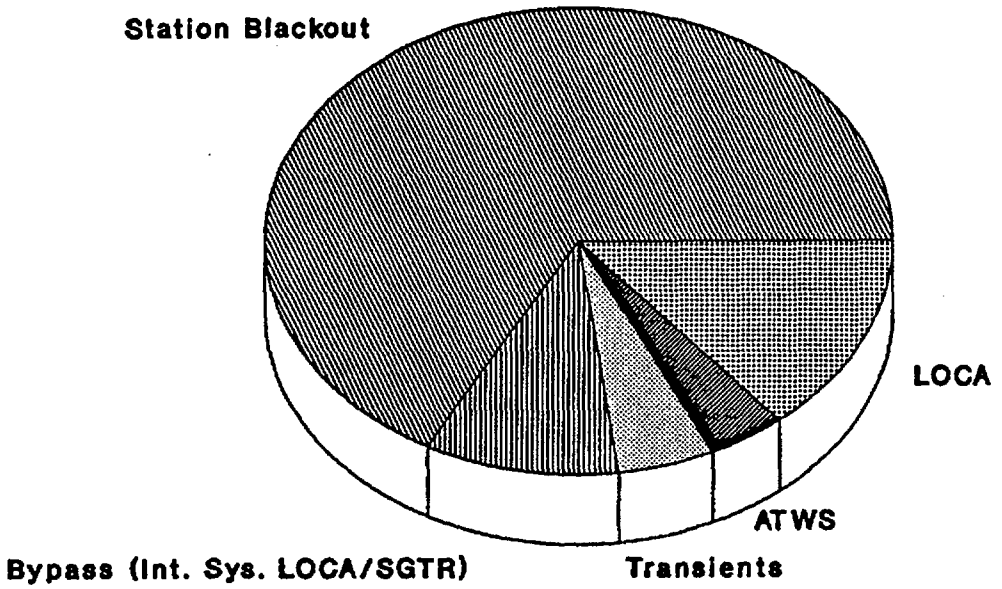


Figure 3.3 Contributors to mean core damage frequency from internal events at Surry.

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:

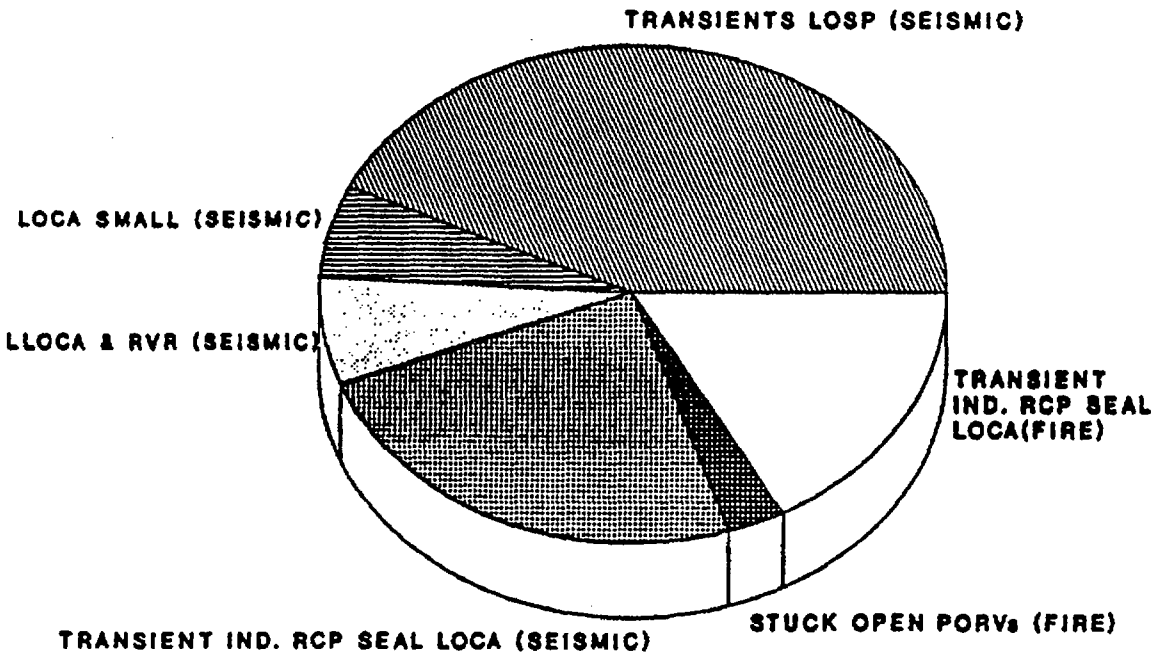
- Seismic, and
- Fire.

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:



Total Mean Core Damage Frequency: 1.3E–4

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.

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

*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.

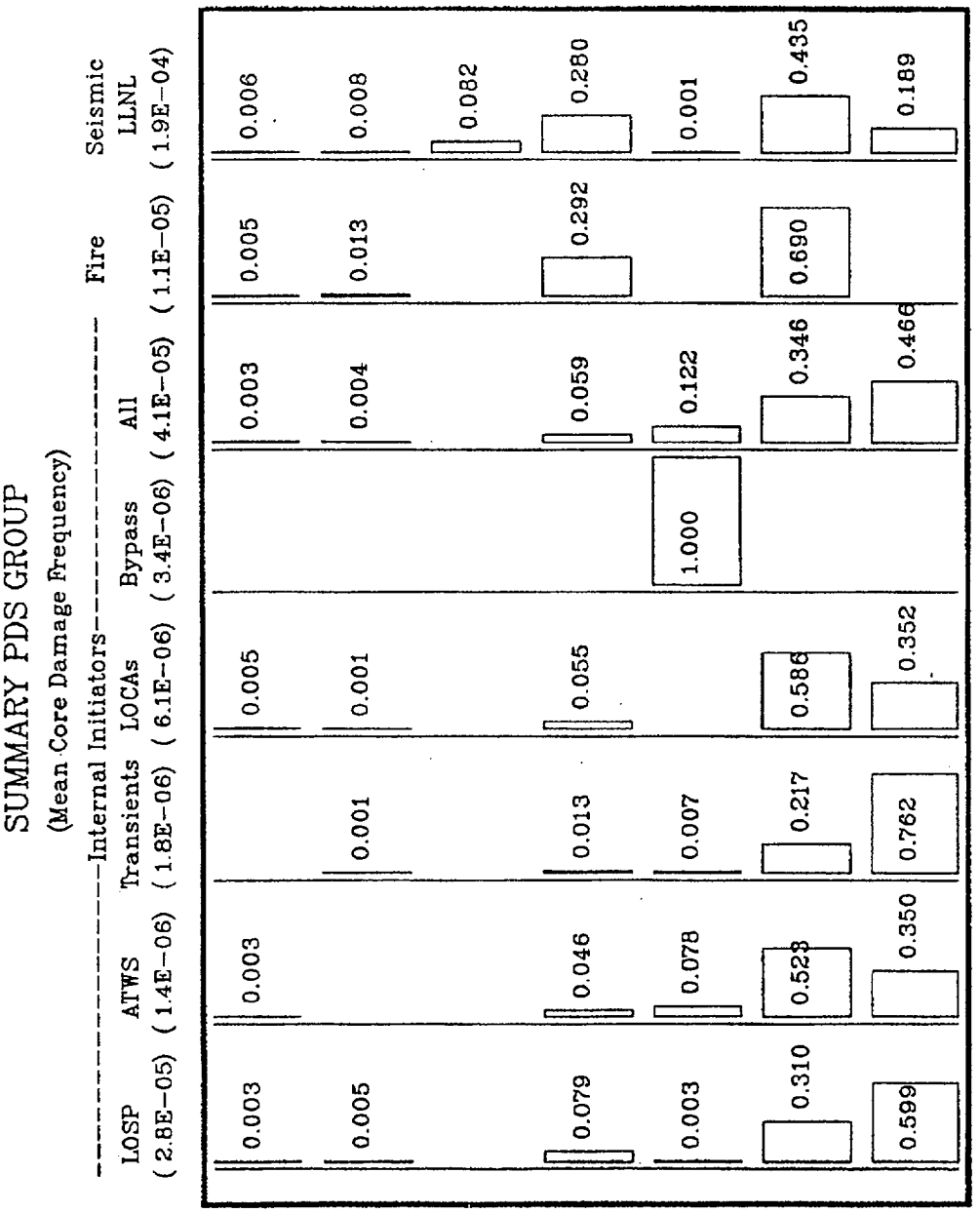
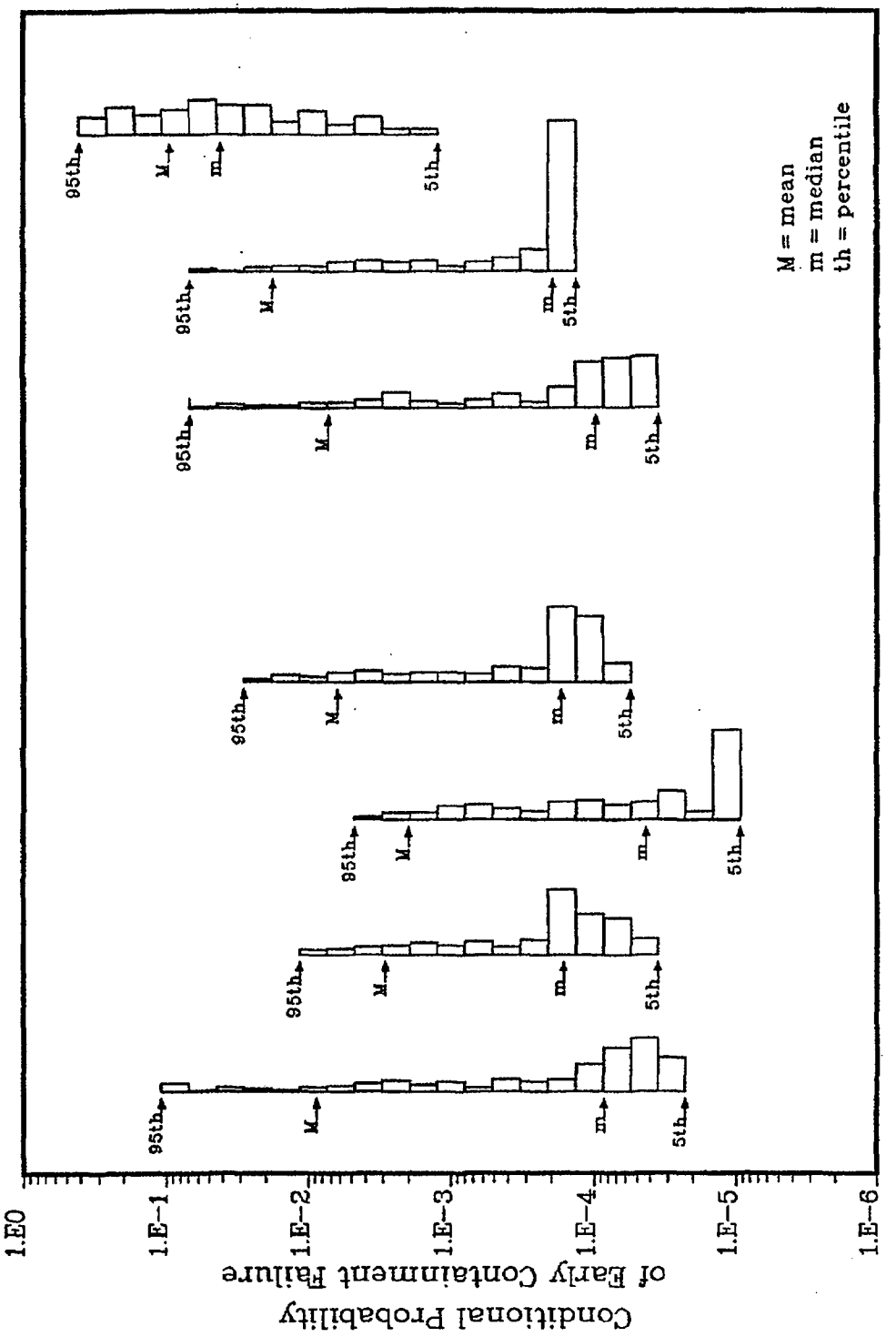


Figure 3.5 Conditional probability of accident progression bins at Surry.



	Internal Initiators					Fire	Seismic
PDS Group	LOSP	ATWS	Transients	LOCAs	Bypass	All	LLNL
Core Damage Freq.	2.8E-05	1.4E-06	1.8E-06	6.1E-06	3.4E-06	4.1E-05	1.1E-05	1.9E-04

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).

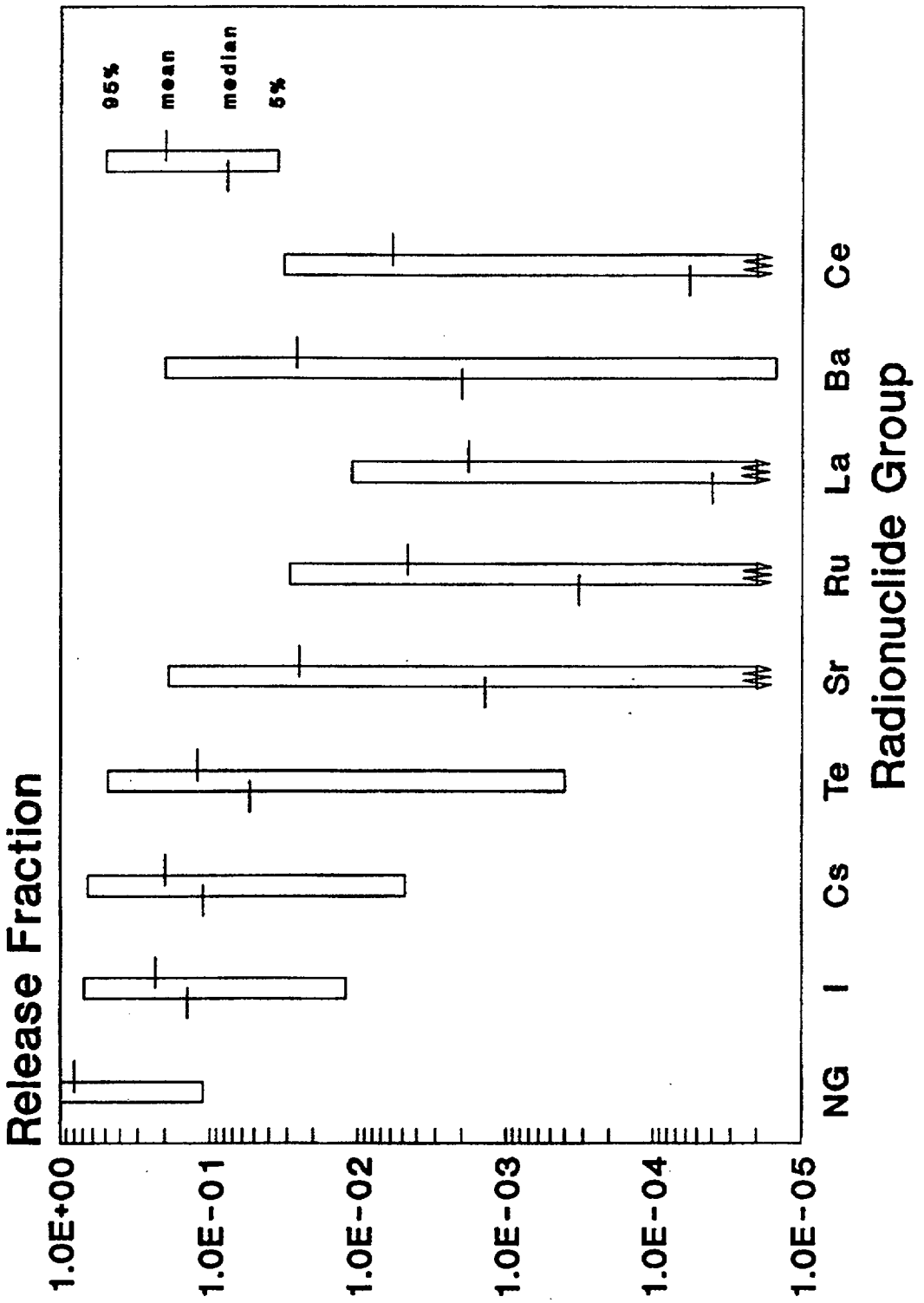


Figure 3.7 Source term distributions for containment bypass at Surry.

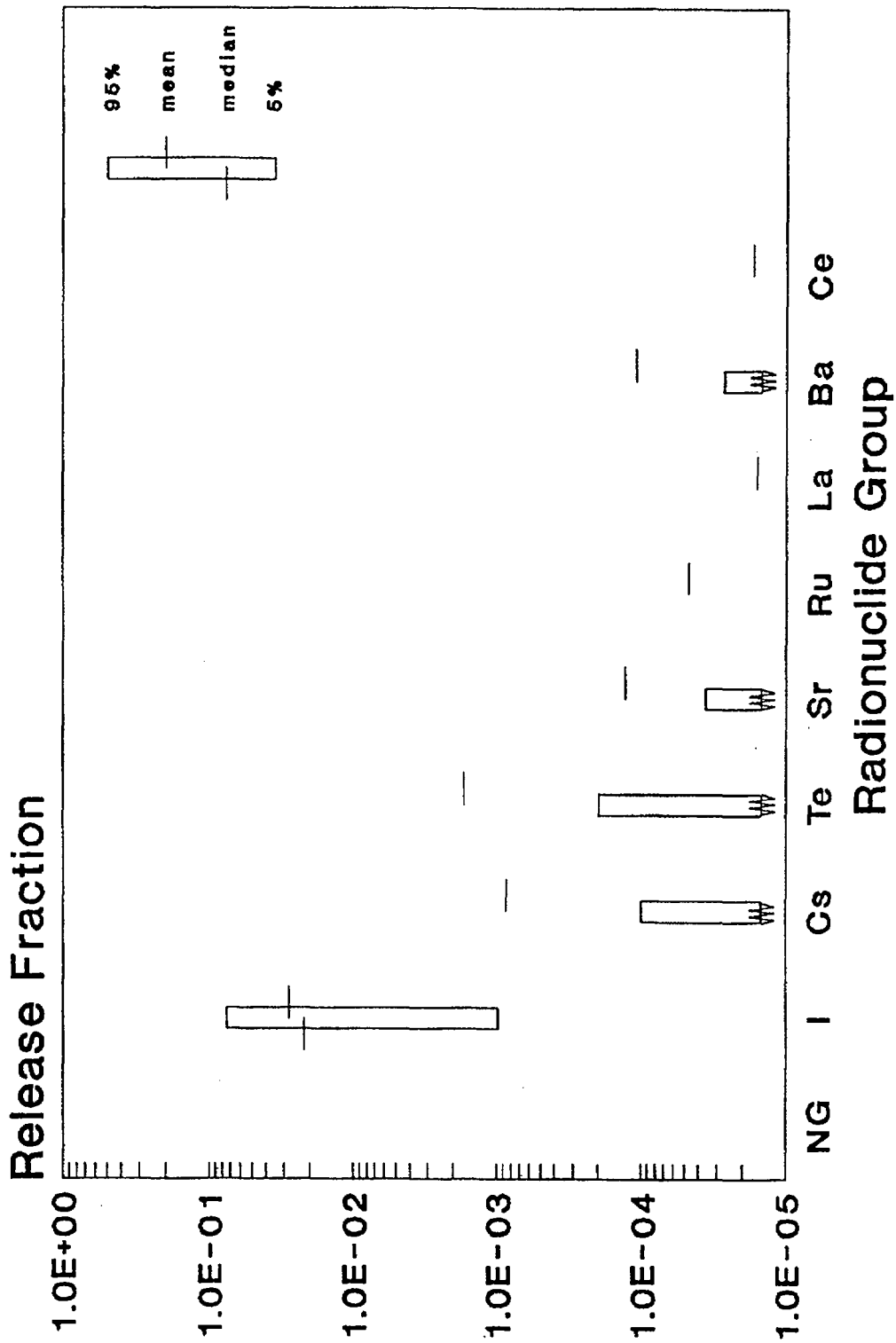


Figure 3.8 Source term distributions for late containment failure at Surry.

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.

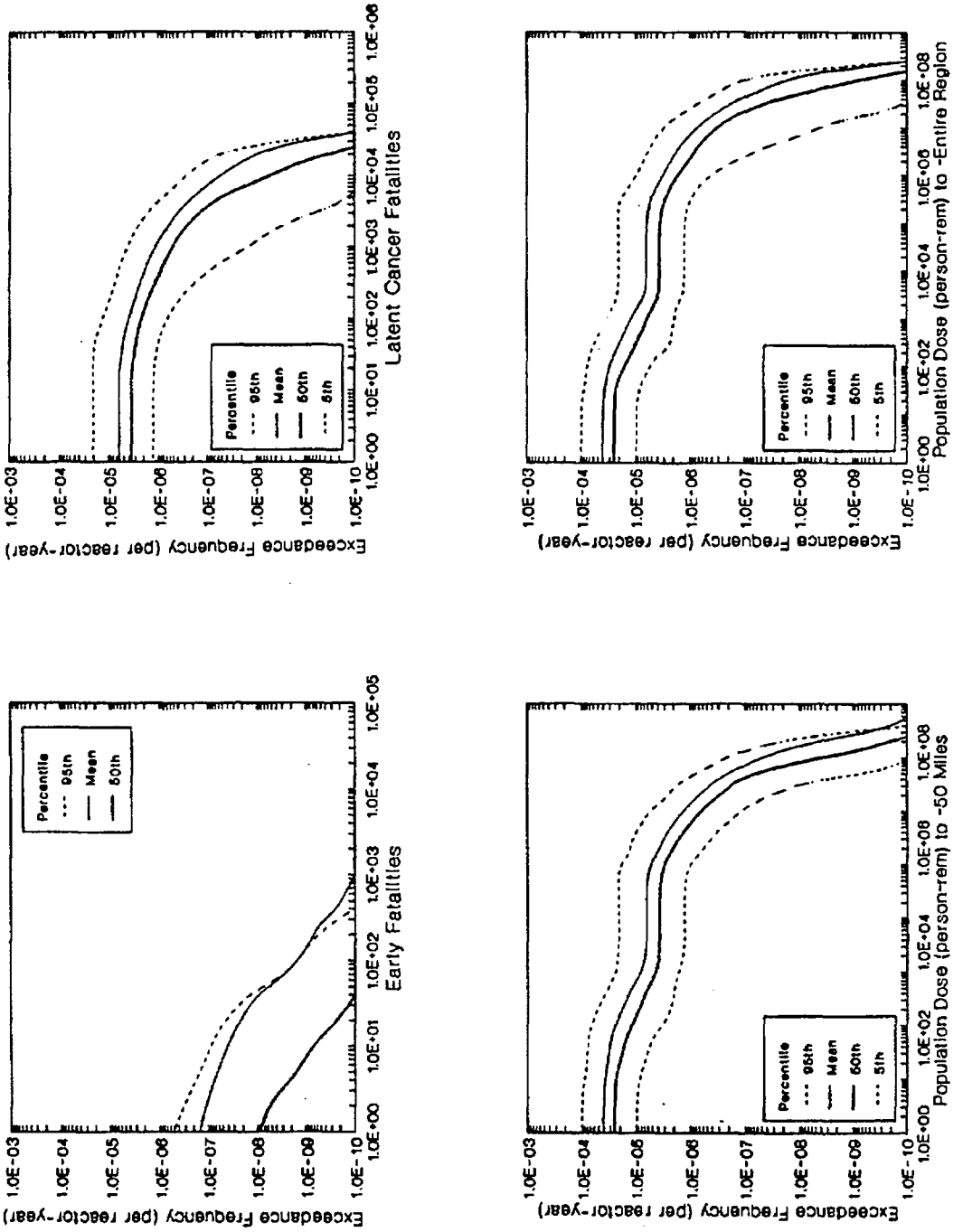
The results displayed in Figures 3.9 and 3.10 are discussed in Chapter 11.

3.6 Public Risk Estimates

3.6.1 Results of Public Risk Estimates

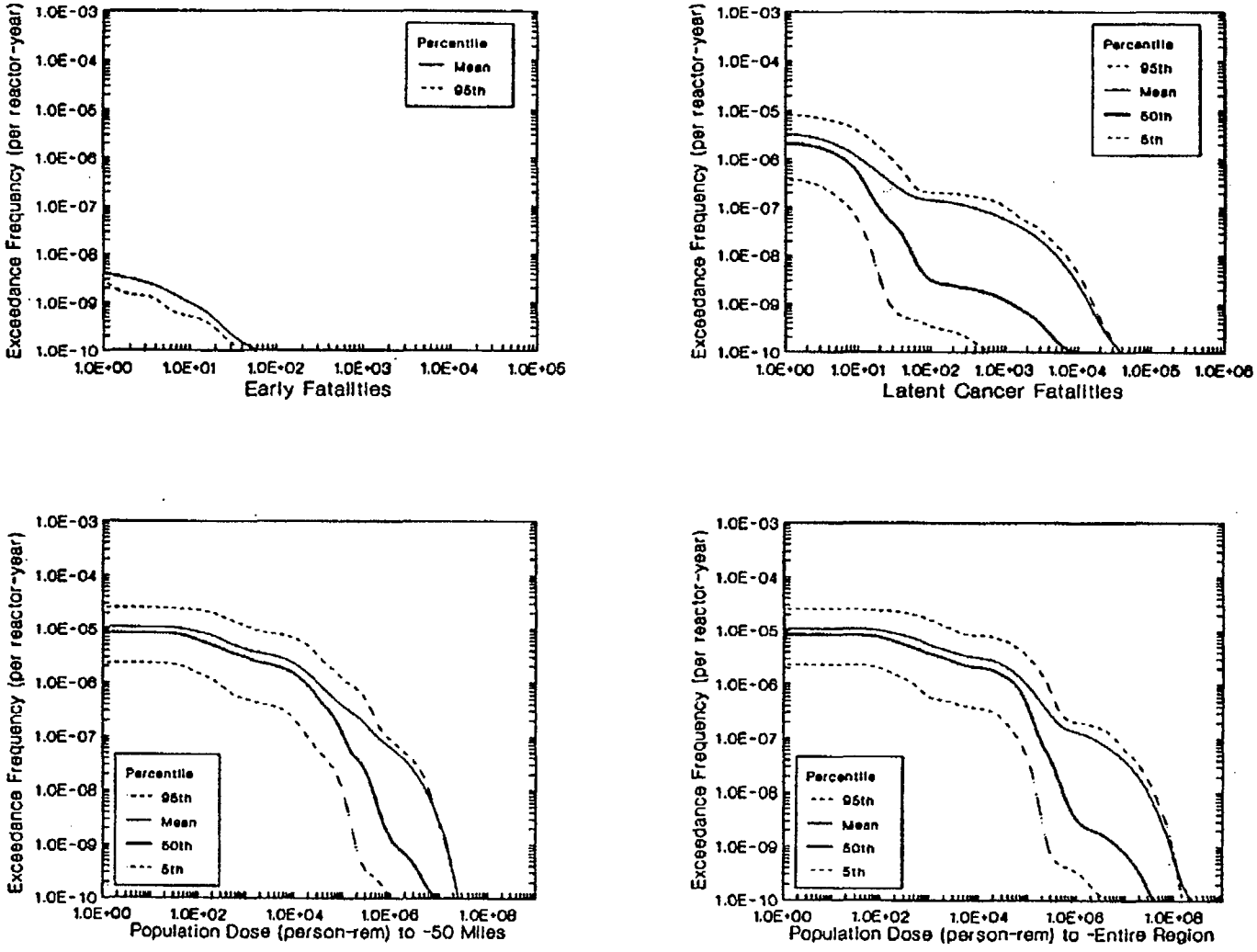
A detailed description of the results of the Surry risk analysis is provided in Reference 3.2. For this summary report, results are provided for the following measures of public risk:

- 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.

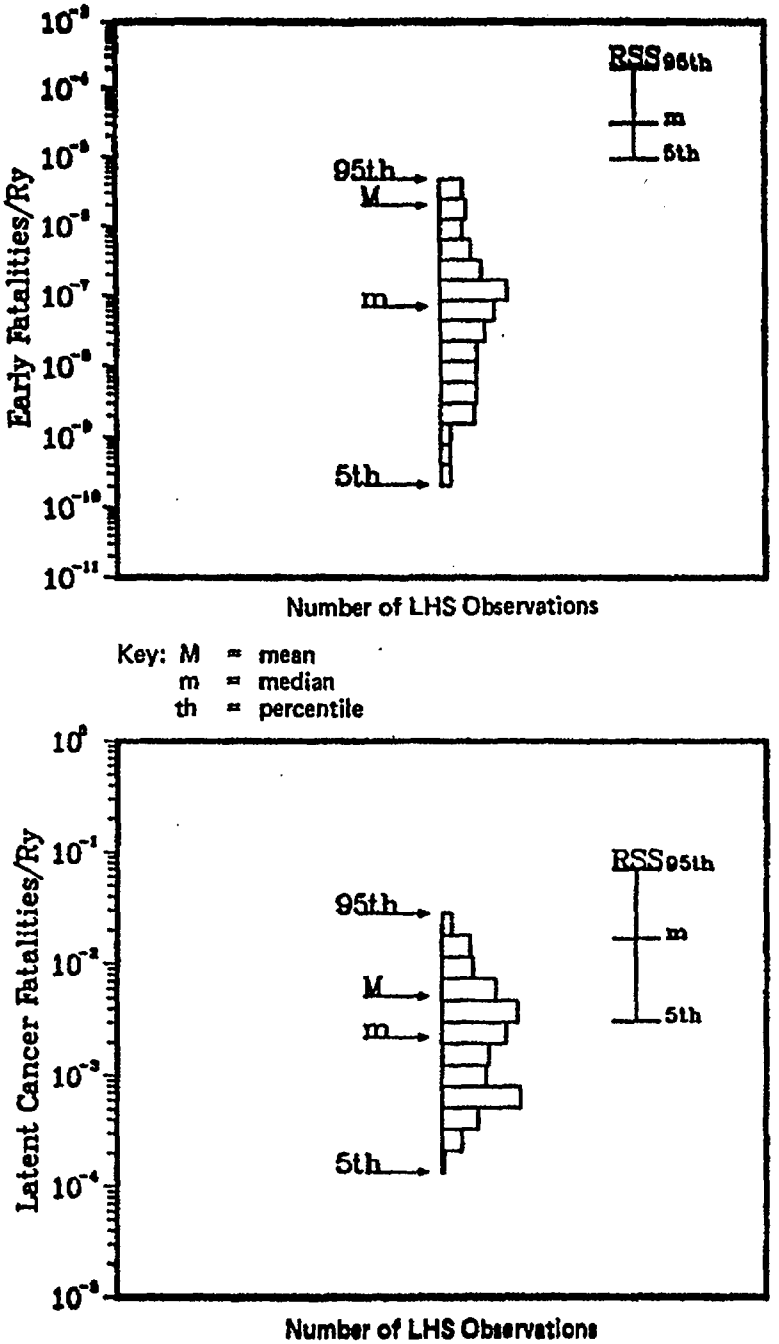
Major contributors to early and latent cancer fatality risks are shown in Figure 3.19. (Note that there are no bypass initiating events in the fire plant damage state.) The most risk-important sequence is a fire in the emergency switchgear room that leads to loss of ac power throughout the station. The 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)

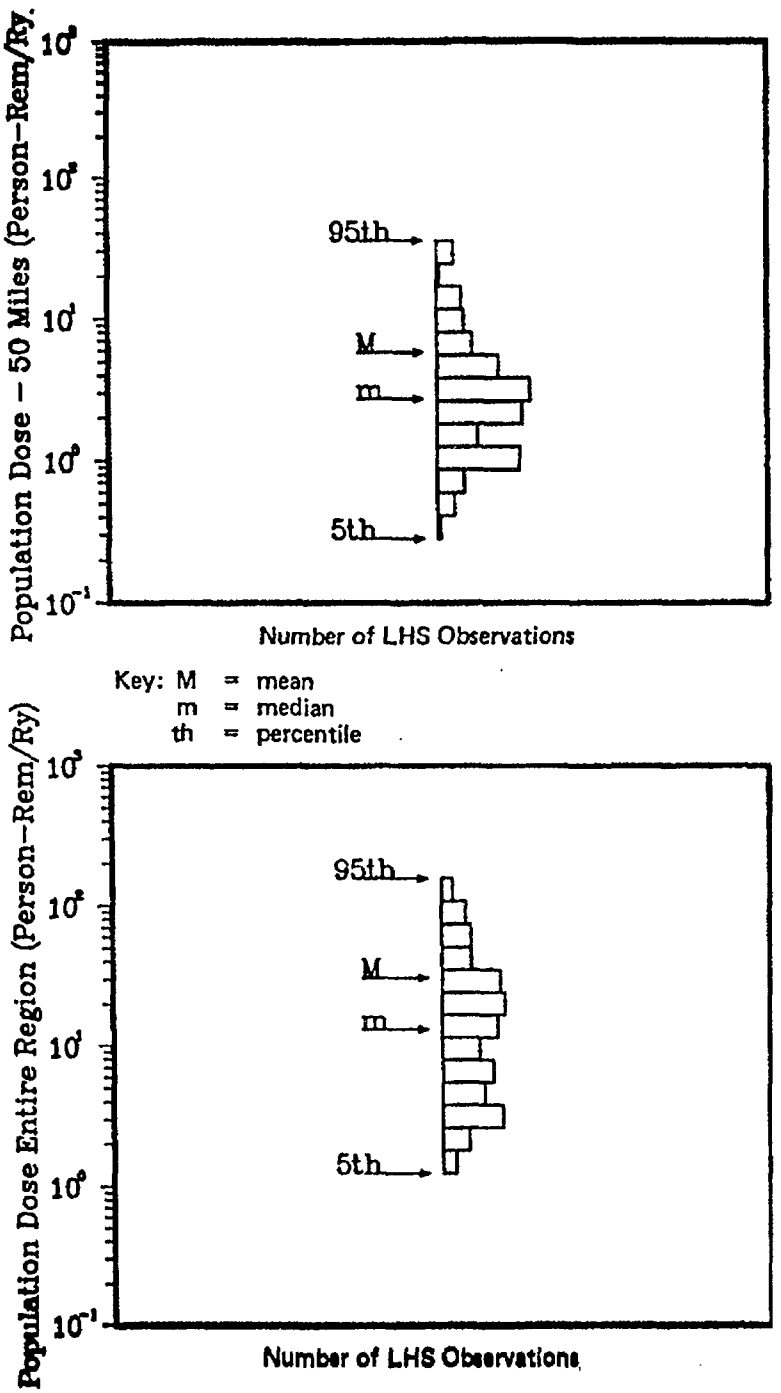
The plant characteristics discussed in Section 3.2.2 that were important in the analysis of core damage frequency were primarily related to the station blackout accident sequences and have not been found to be important in the risk analysis.

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



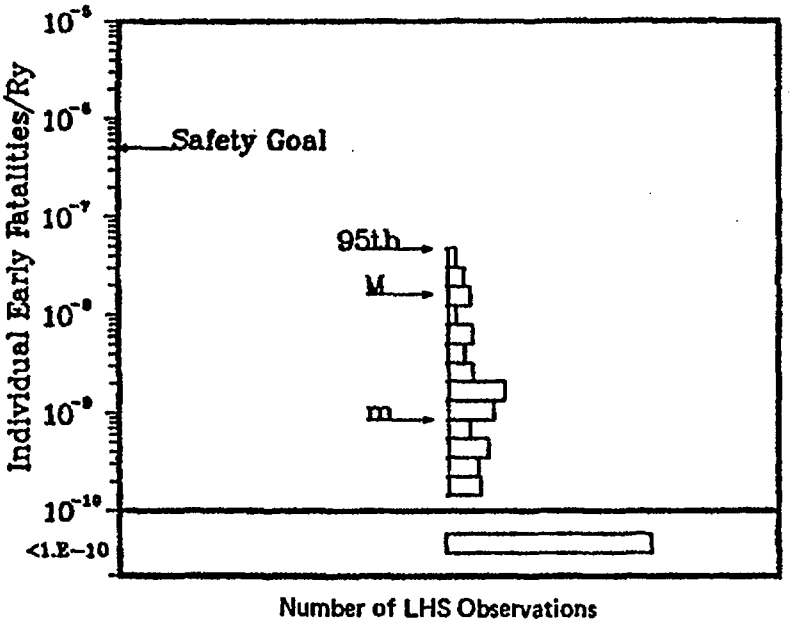
Note: As discussed in Reference 3.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

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

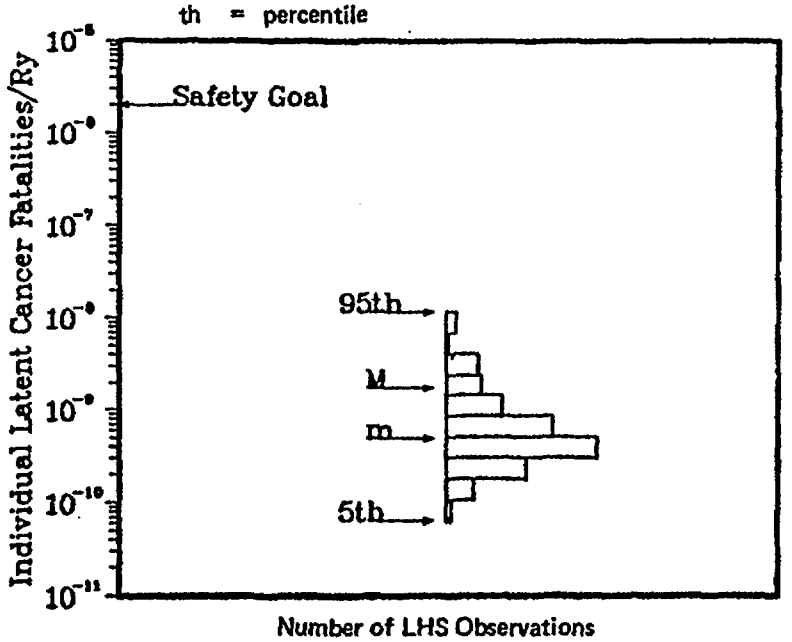


Note: As discussed in Reference 3.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 3.12 Population dose risks at Surry (internal initiators).



Key: M = mean
m = median
th = percentile



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).

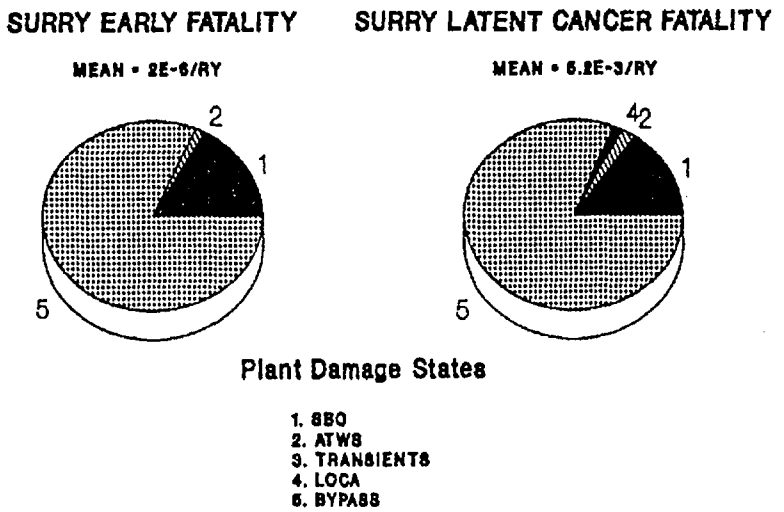


Figure 3.14 Major contributors (plant damage states) to mean early and latent cancer fatality risks at Surry (internal initiators).

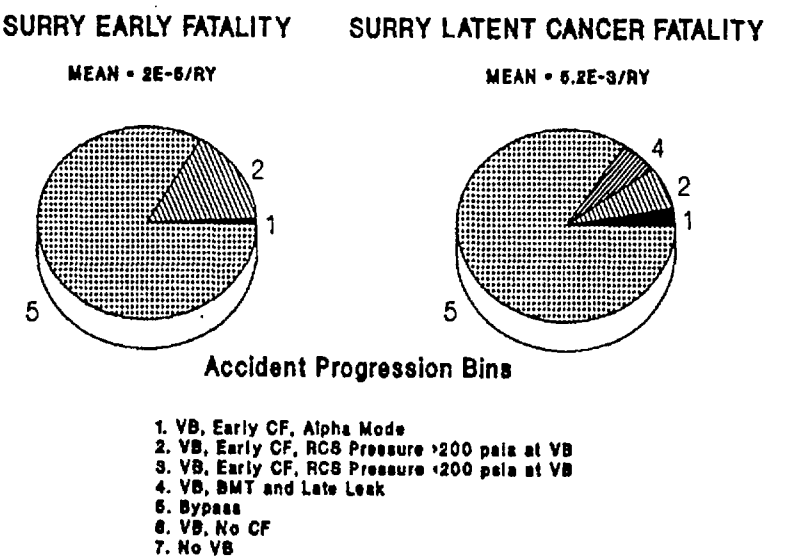
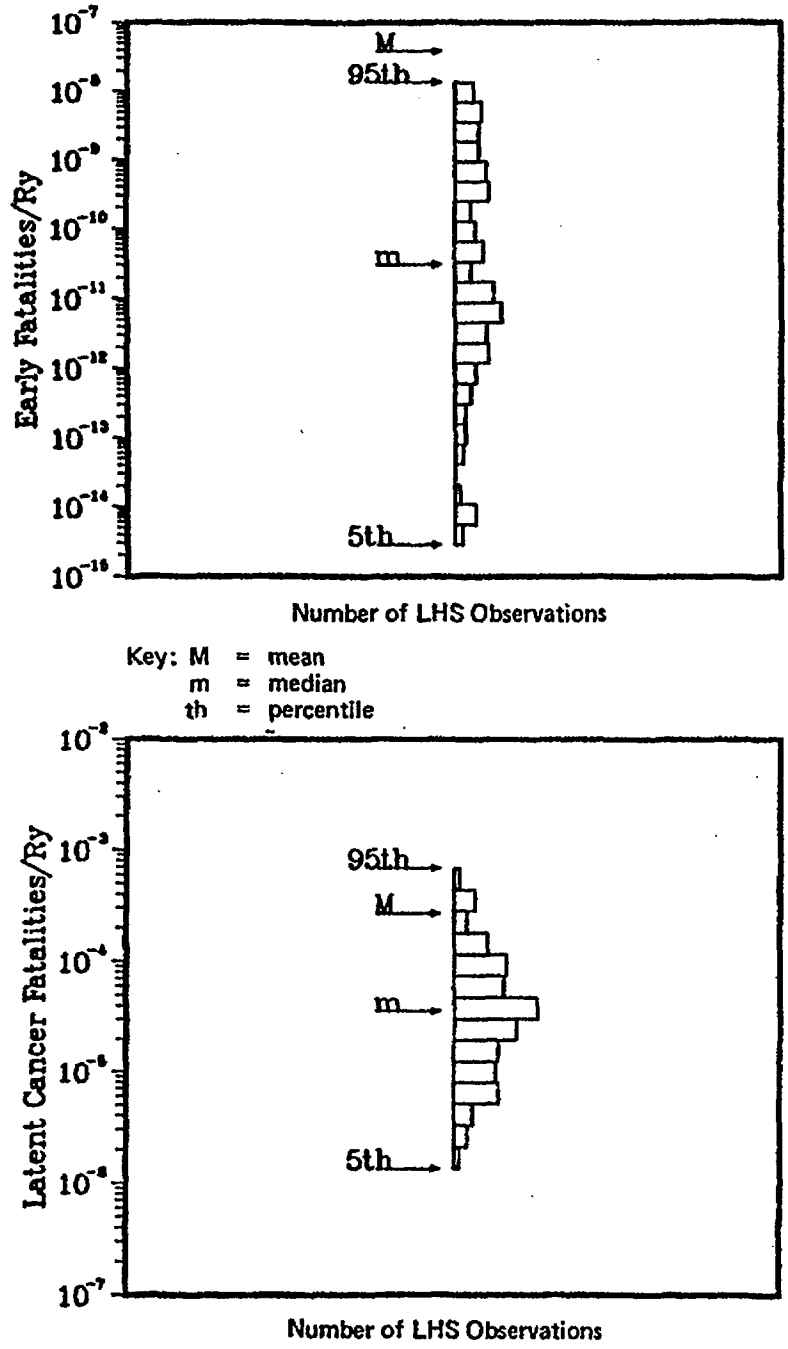
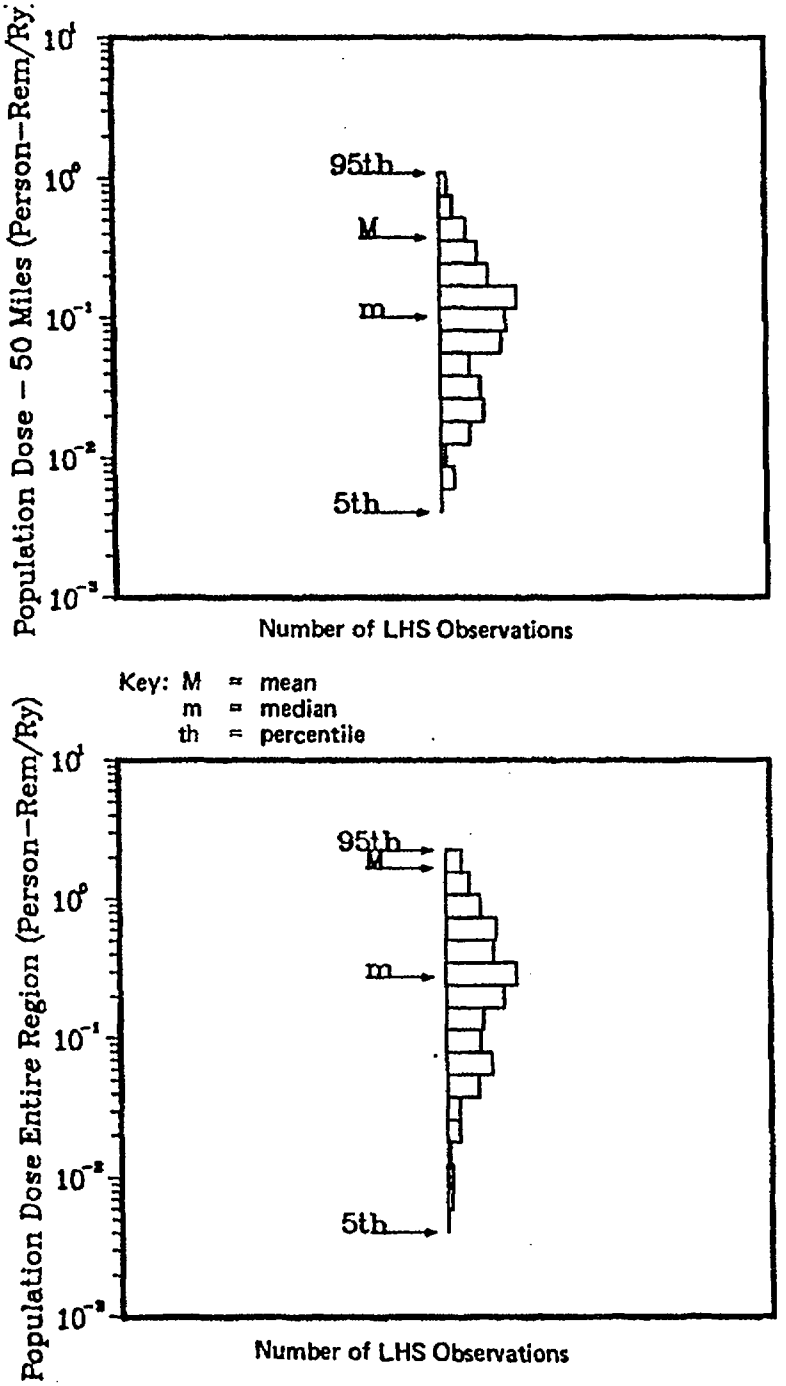


Figure 3.15 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Surry (internal initiators).



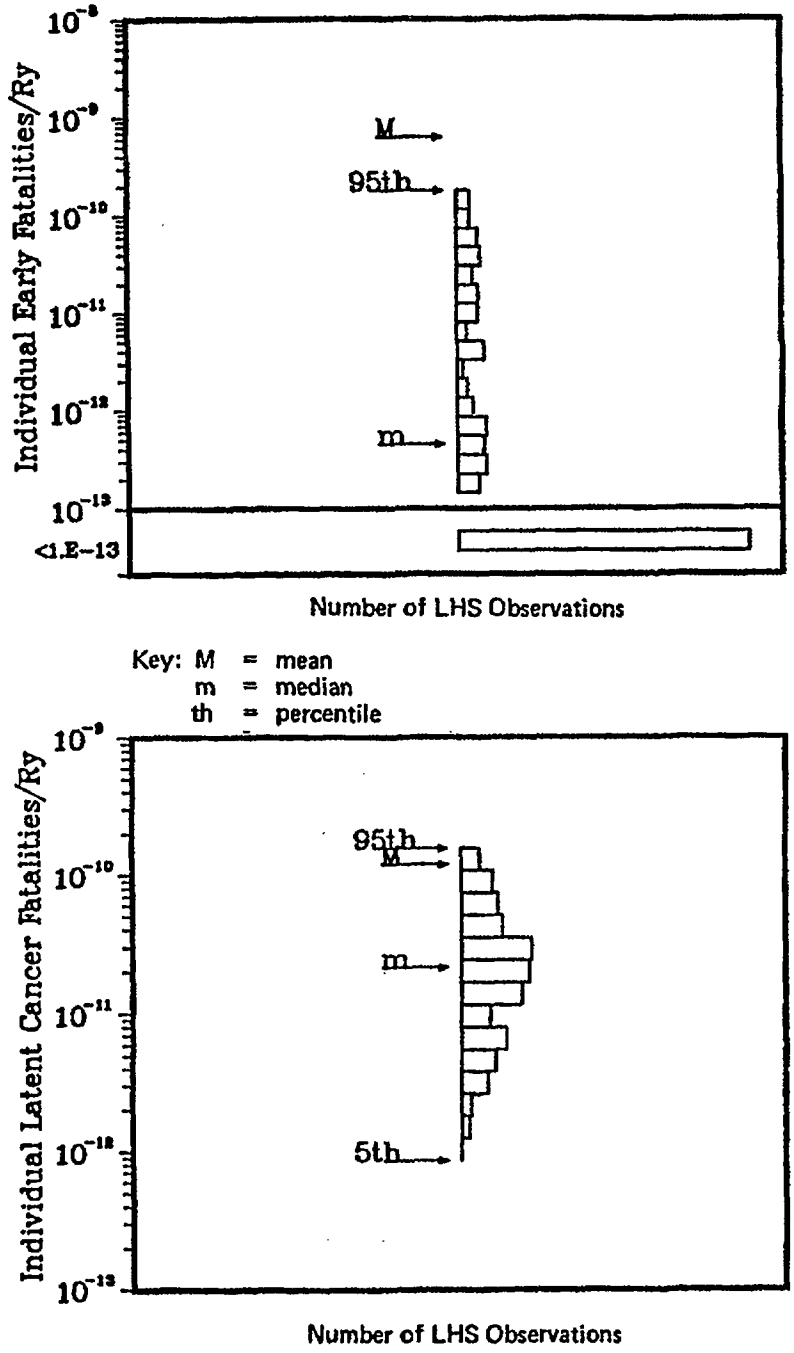
Note: As discussed in Reference 3.4, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

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).

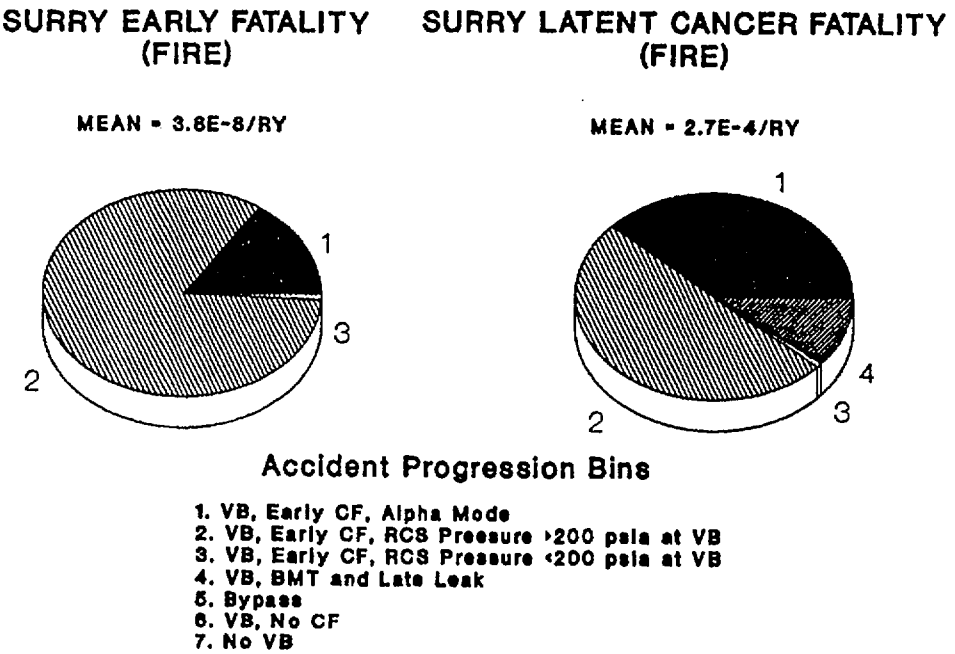


Figure 3.19 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Surry (fire initiators).

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.

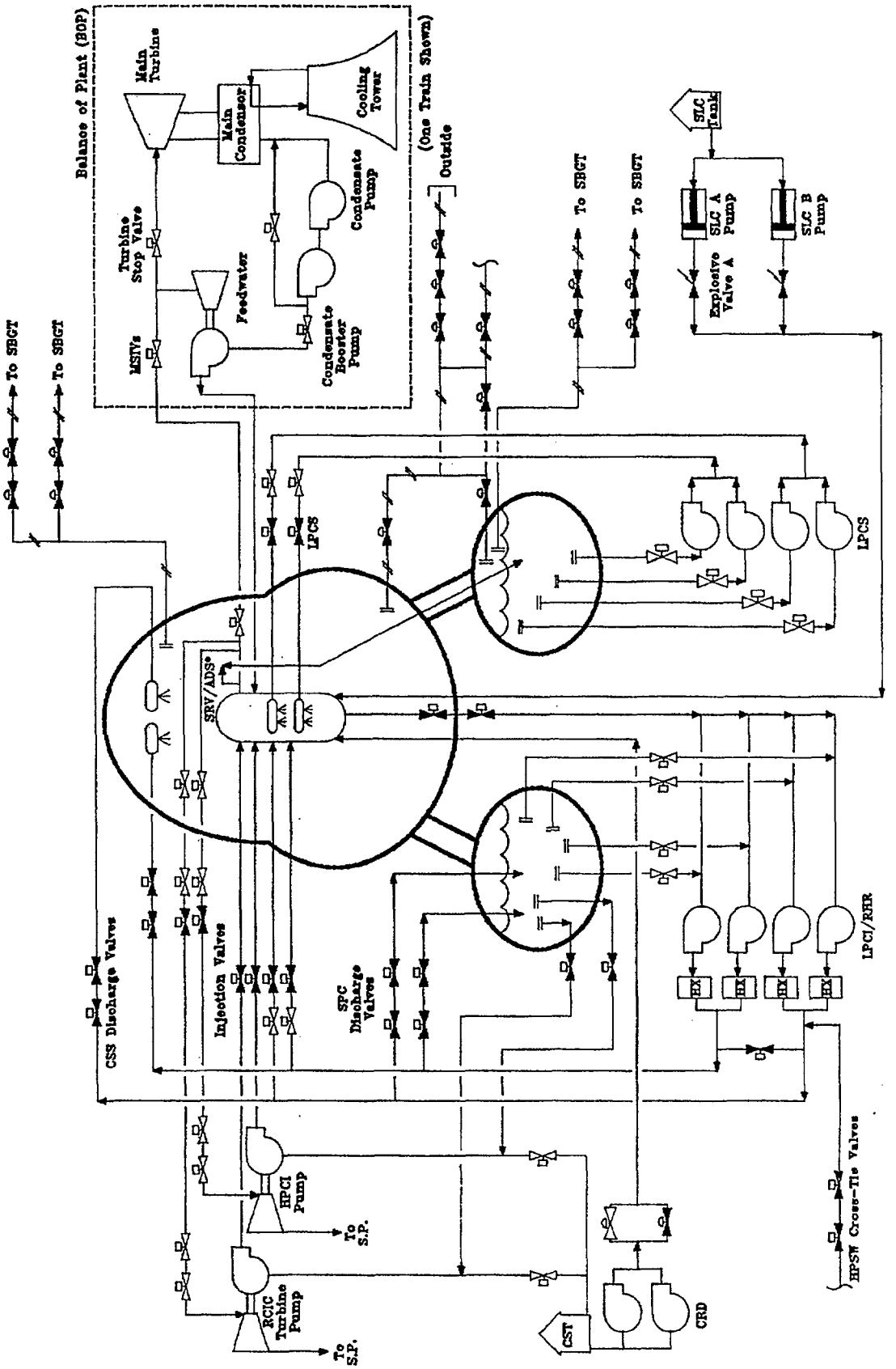
The relative contributions of these groups to mean internal-event core damage frequency at Peach Bottom are shown in Figure 4.3. From Figure 4.3, it may be seen that station blackout sequences as a class are the largest contributor to mean core damage frequency. It should be noted that the plant configuration (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

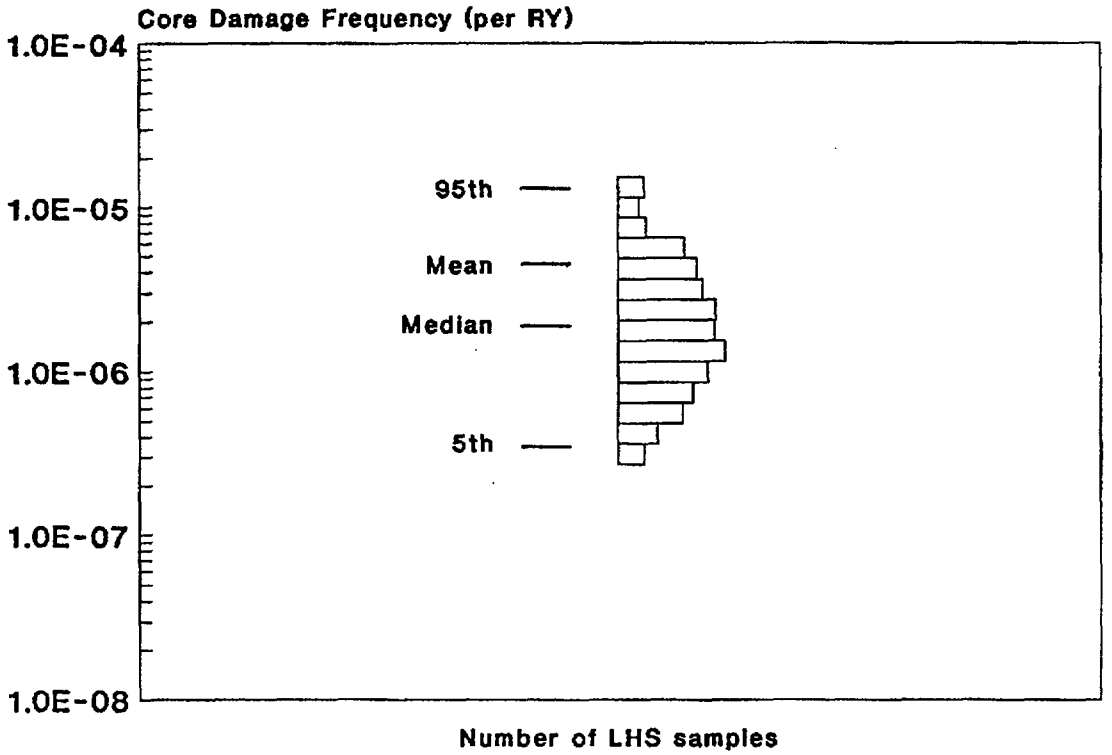
Table 4.1 Summary of design features: Peach Bottom Unit 2.

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).

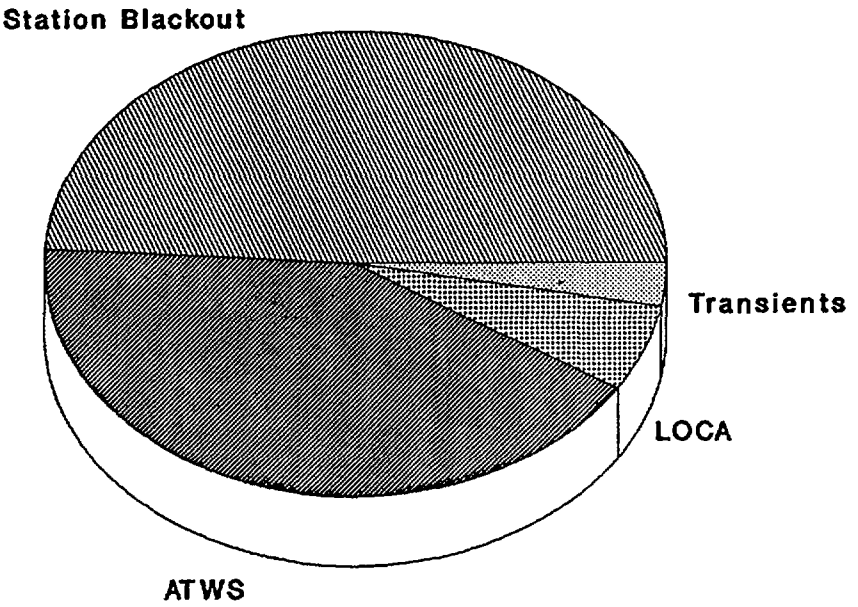
Figure 4.2 Internal core damage frequency results at Peach Bottom.

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.

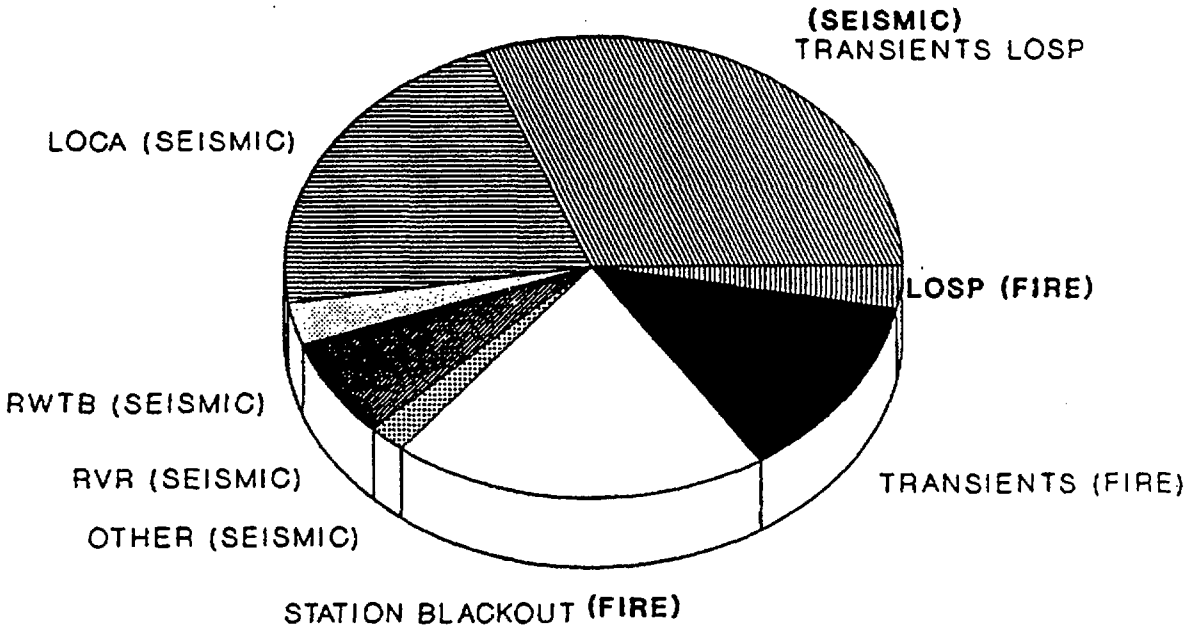
Fires in the cable-spreading room are assumed to require manual plant trip and to fail the high-pressure injection and depressurization systems, namely: high pressure core injection (HPCI), reactor core isolation cooling (RCIC), control rod drive (CRD), and automatic depressurization systems (ADS). In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



Total Mean Core Damage Frequency: 9.7E-5

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.

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.

*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.

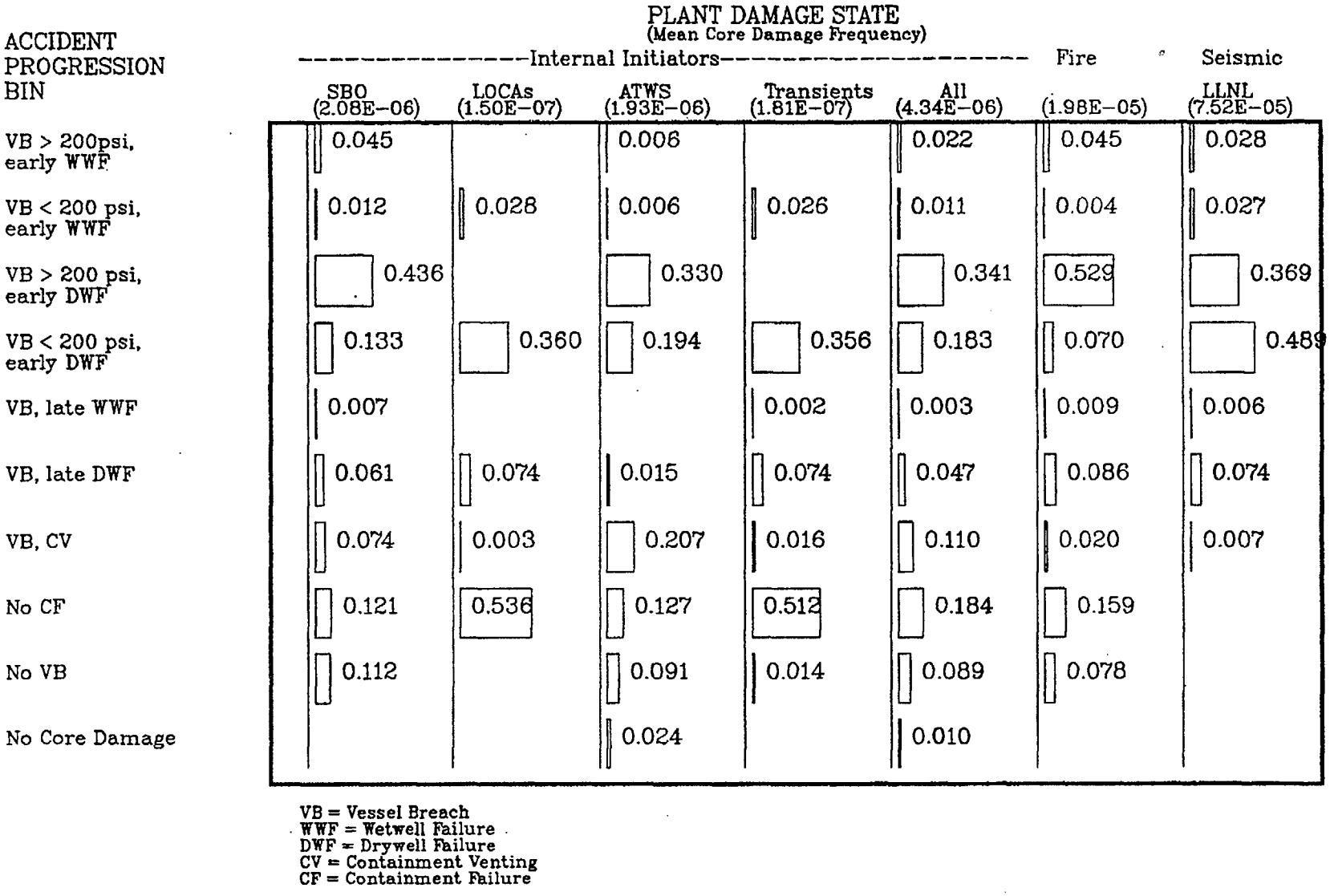


Figure 4.5 Conditional probability of accident progression bins at Peach Bottom.

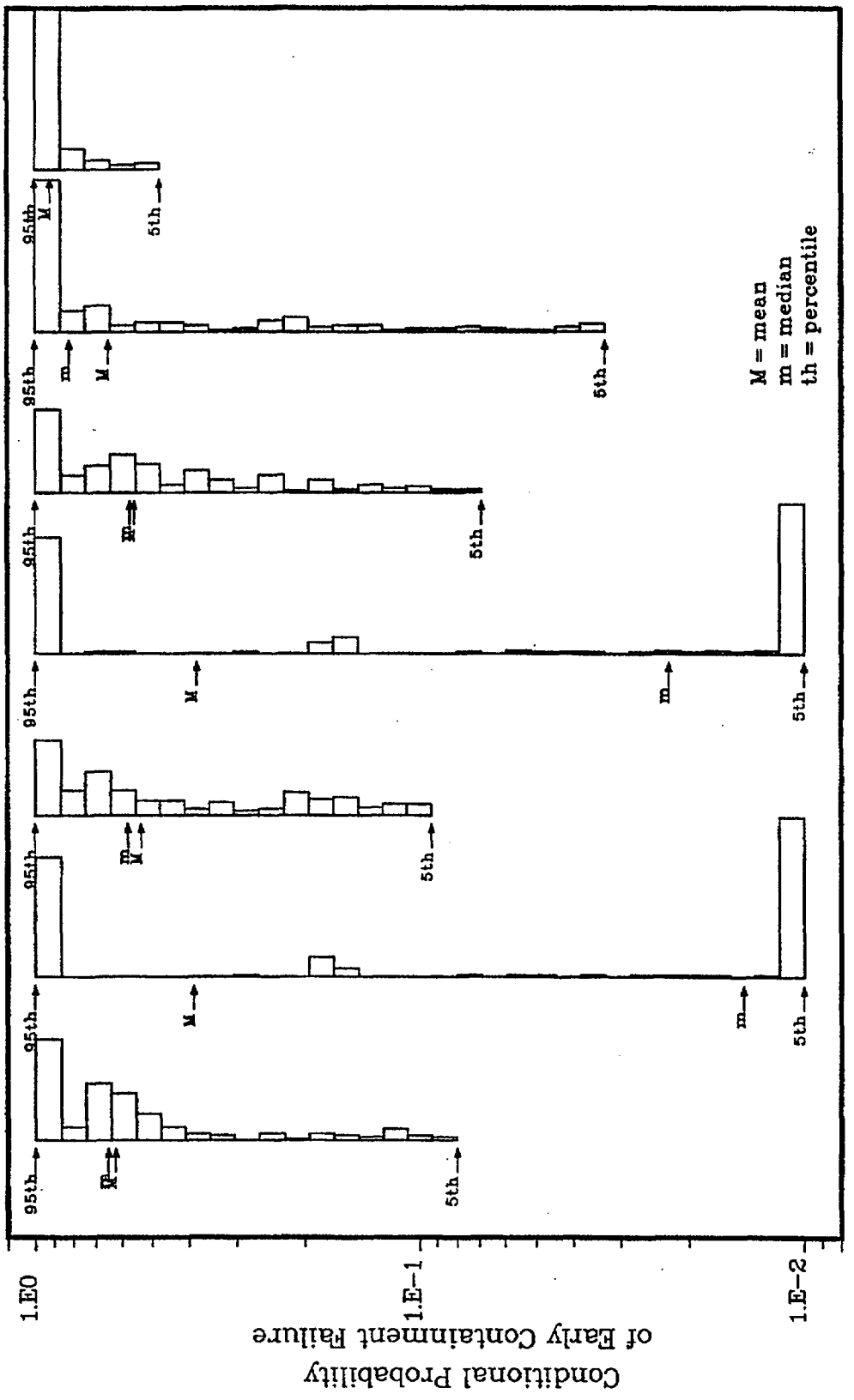


Figure 4.6 Conditional probability distributions for early containment failure at Peach Bottom.

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.

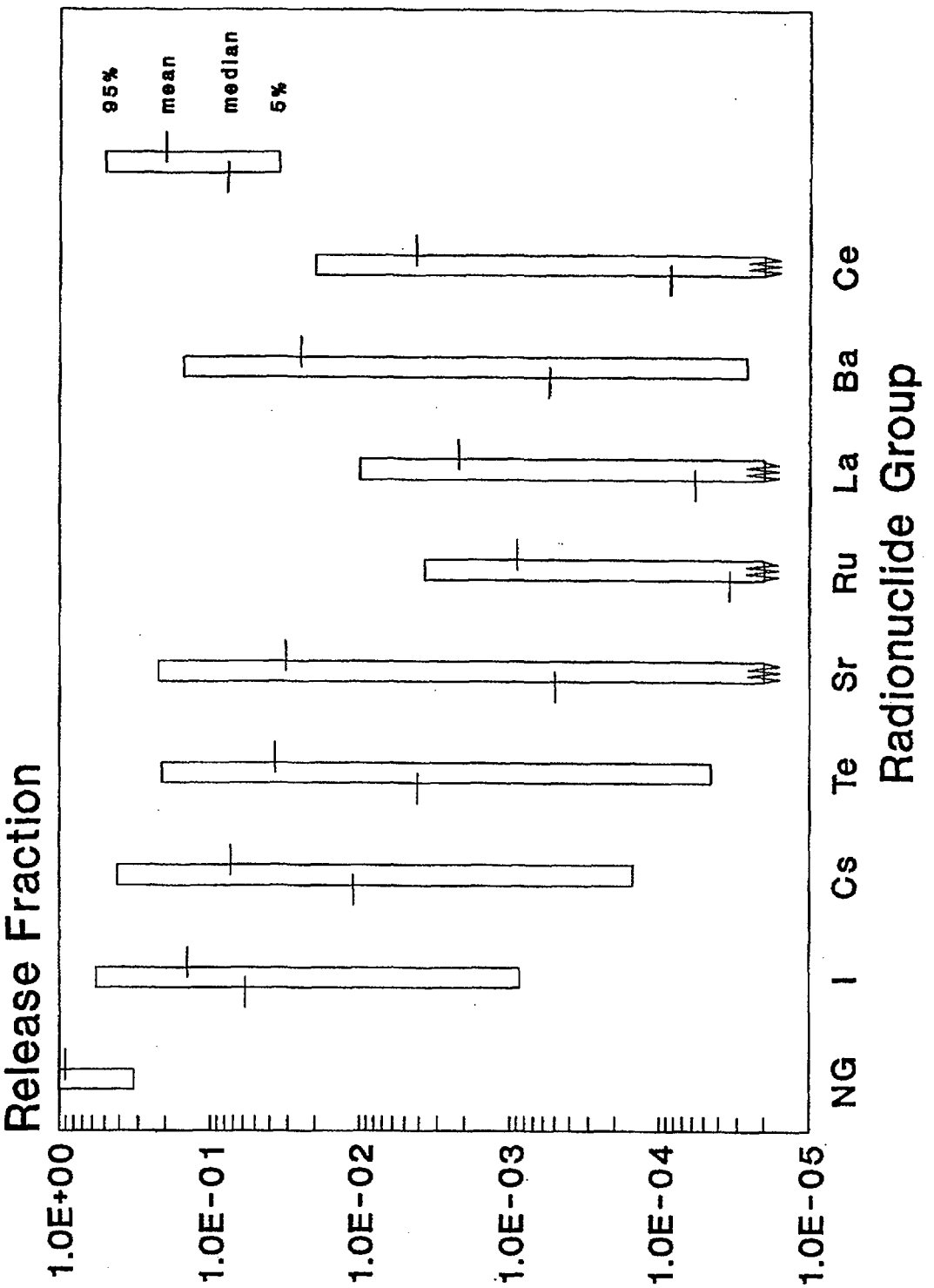


Figure 4.7 Source term distributions for early failure in drywell at Peach Bottom.

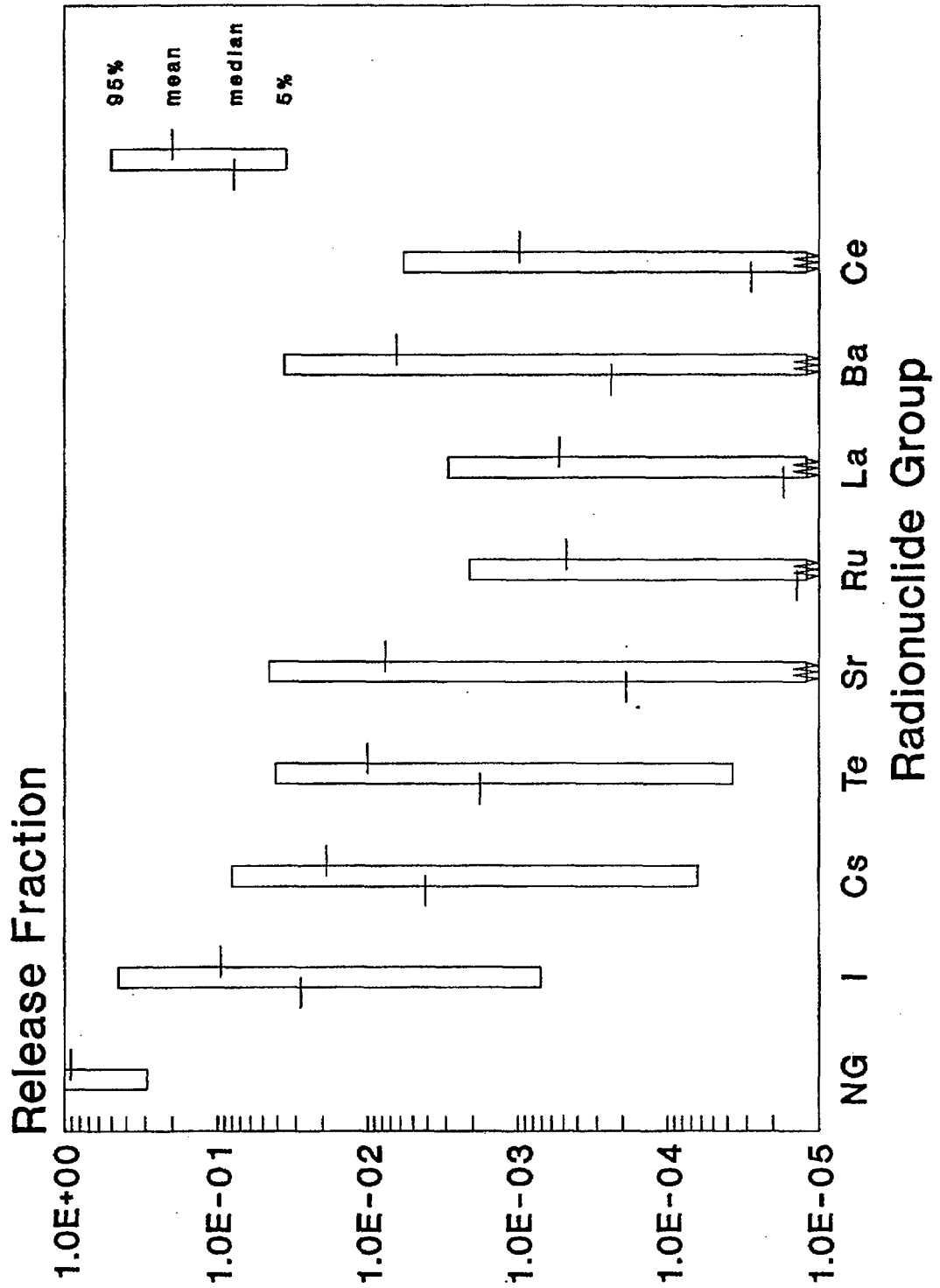


Figure 4.8 Source term distributions for vented containment at Peach Bottom.

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.

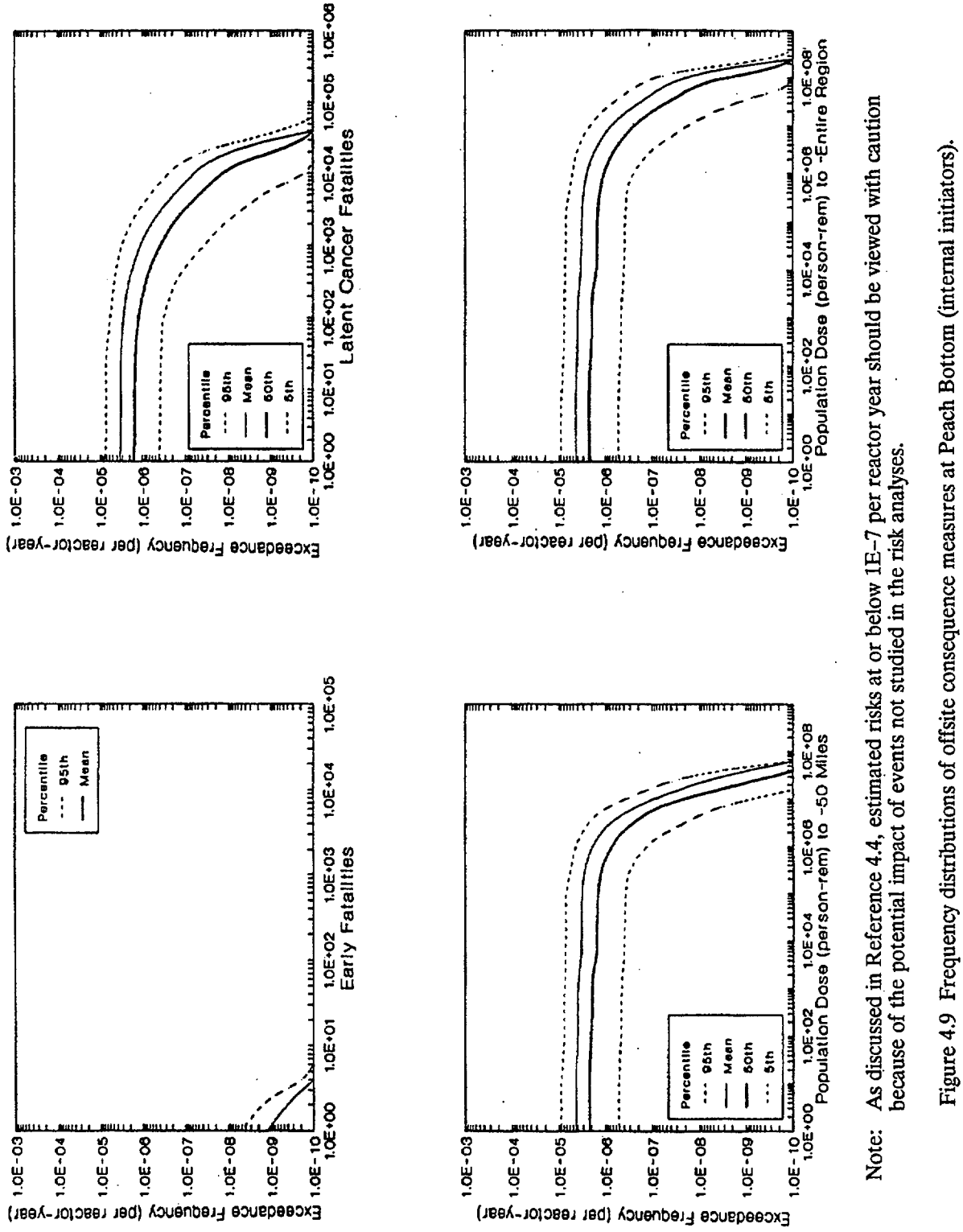
The results displayed in Figures 4.9 and 4.10 are discussed in Chapter 11.

4.6 Public Risk Estimates

4.6.1 Results of Public Risk Estimates

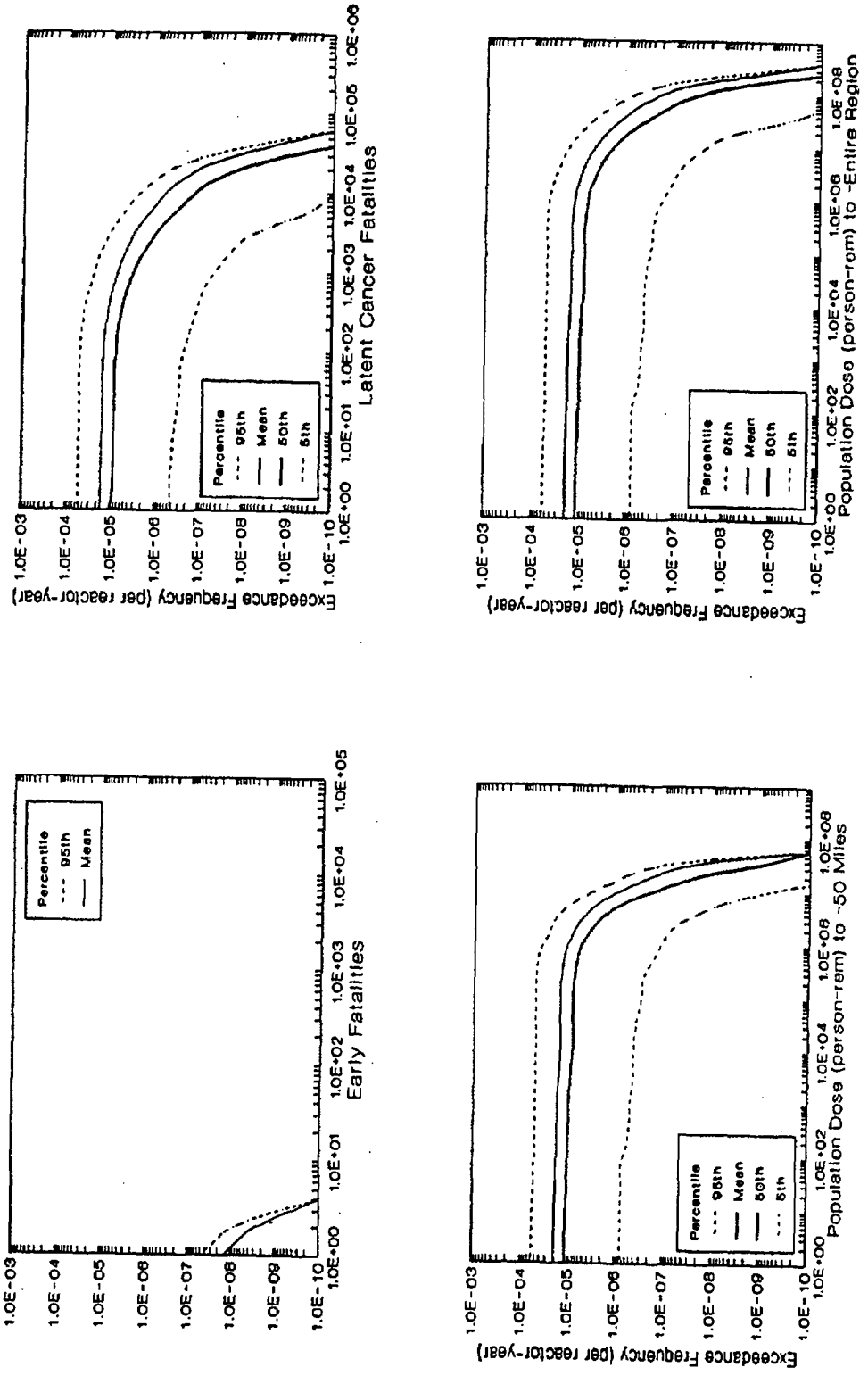
A detailed description of the results of the Peach Bottom risk is provided in Reference 4.2. For this summary report, results are provided for the following measures of public risk:

- 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).

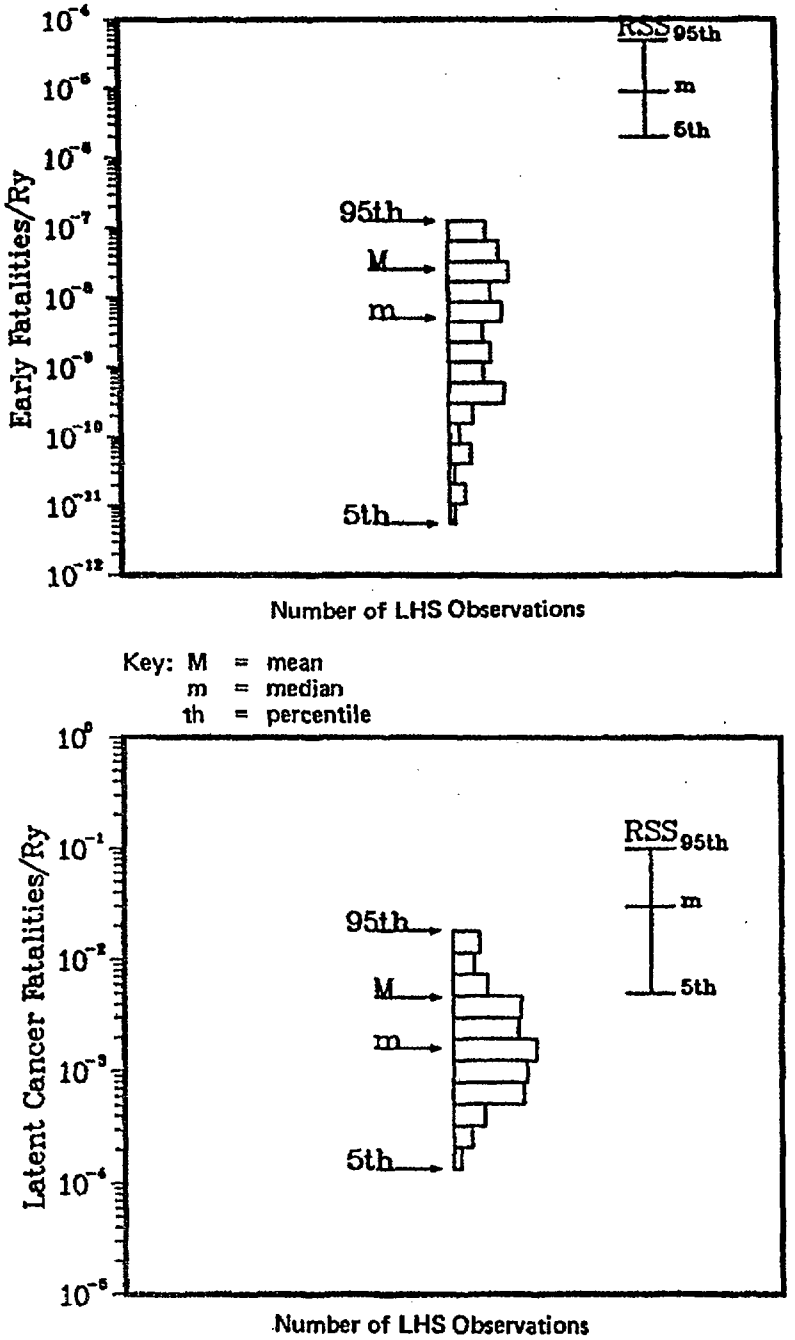
4.6.1.2 Externally Initiated Accident Sequences

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.

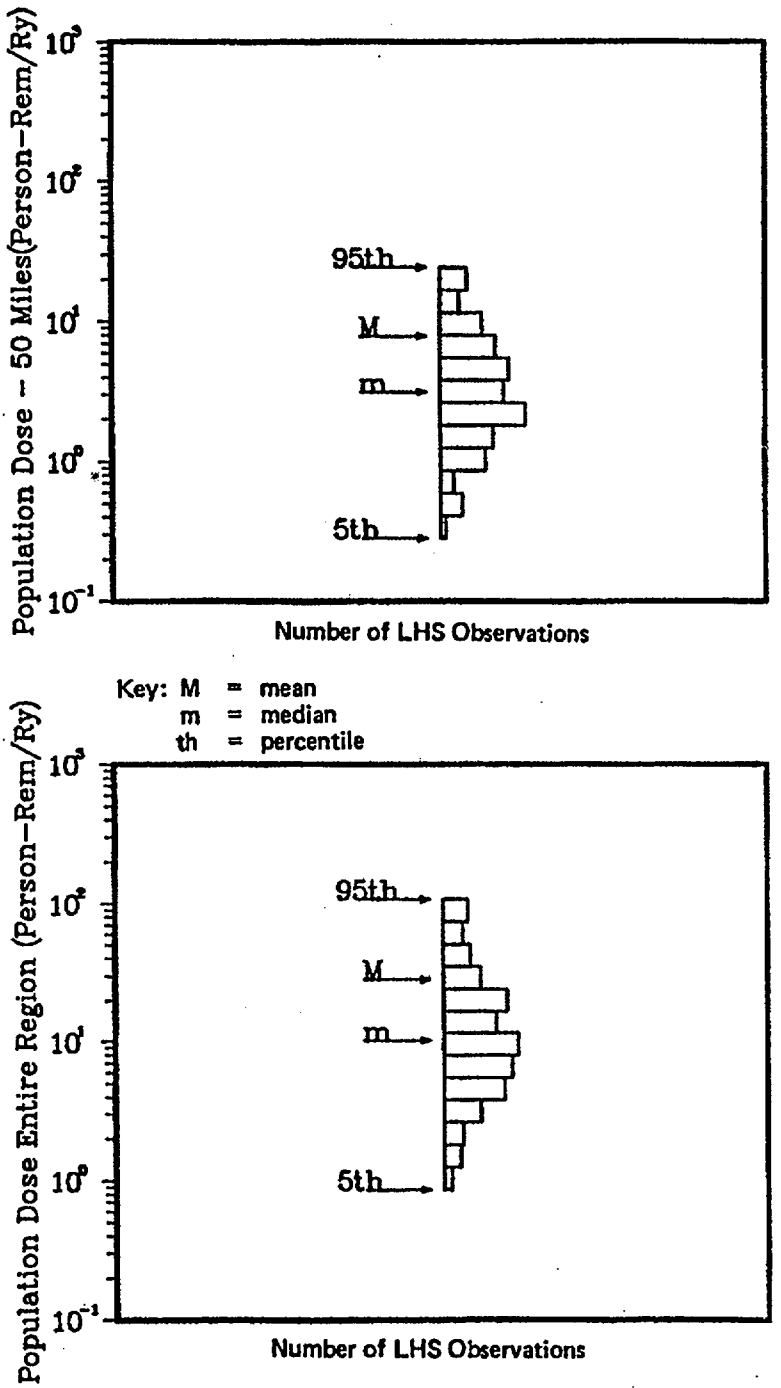
4.6.2 Important Plant Characteristics (Risk)

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.



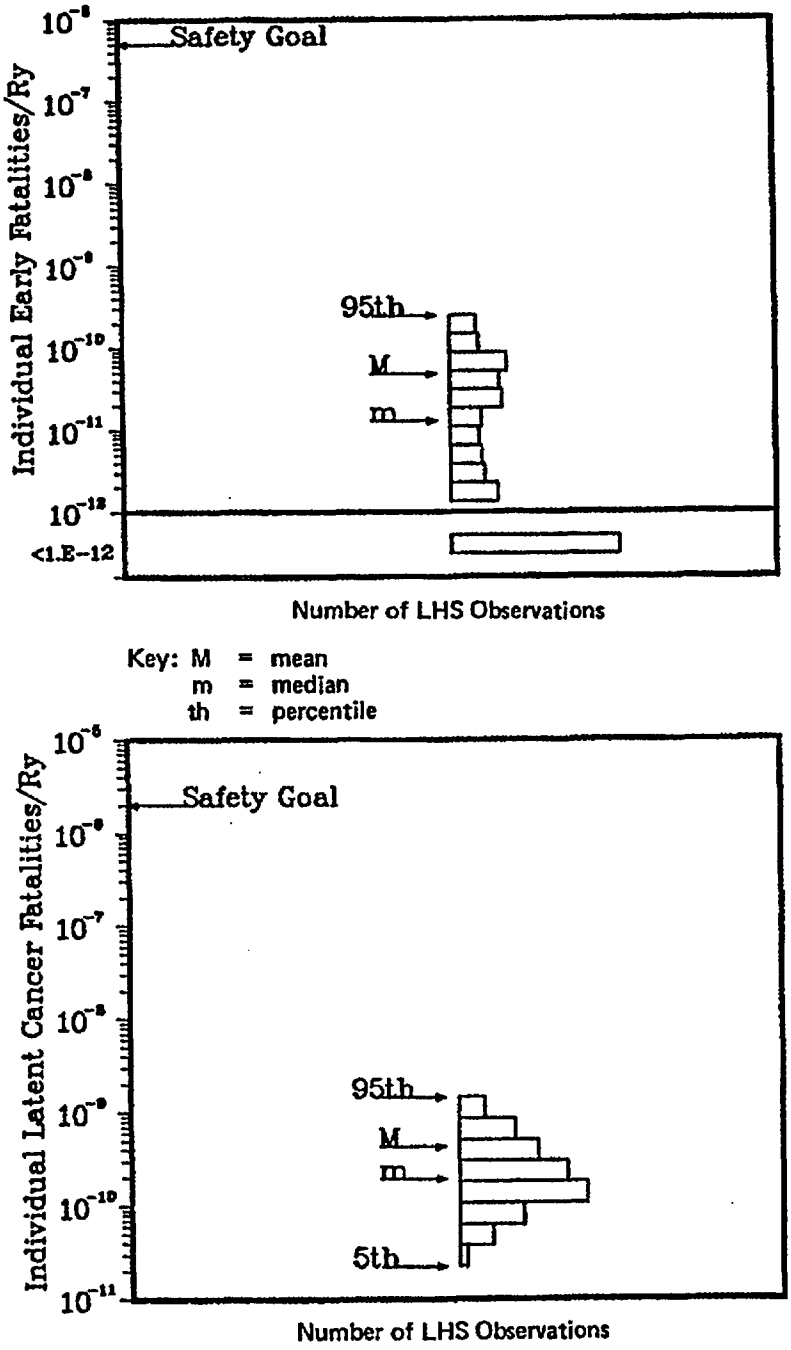
Note: As discussed in Reference 4.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

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).

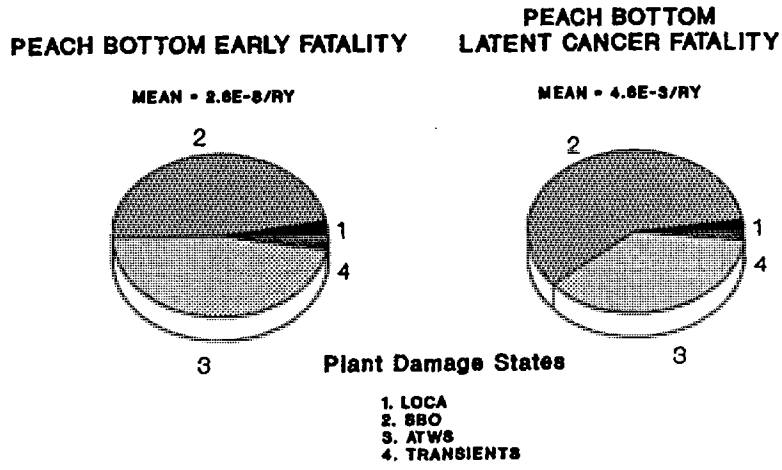


Figure 4.14 Major contributors (plant damage states) to mean early and latent cancer fatality risks at Peach Bottom (internal initiators).

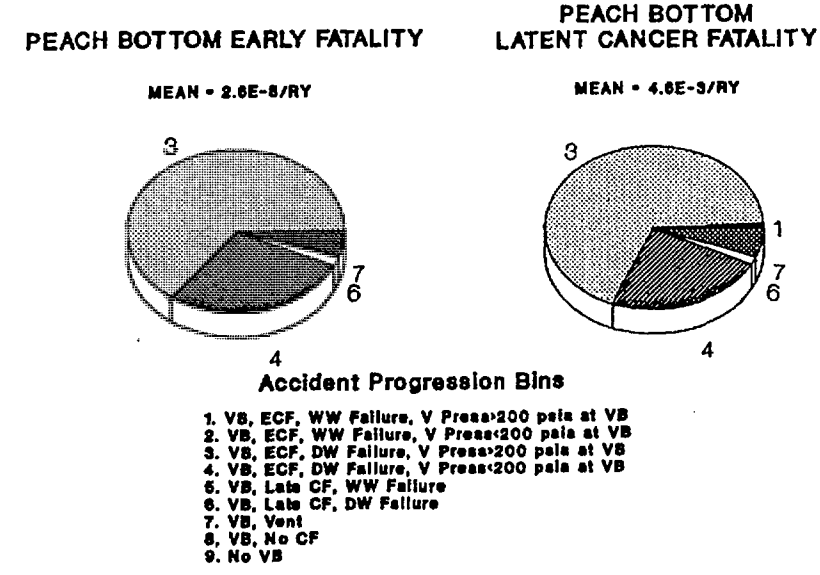
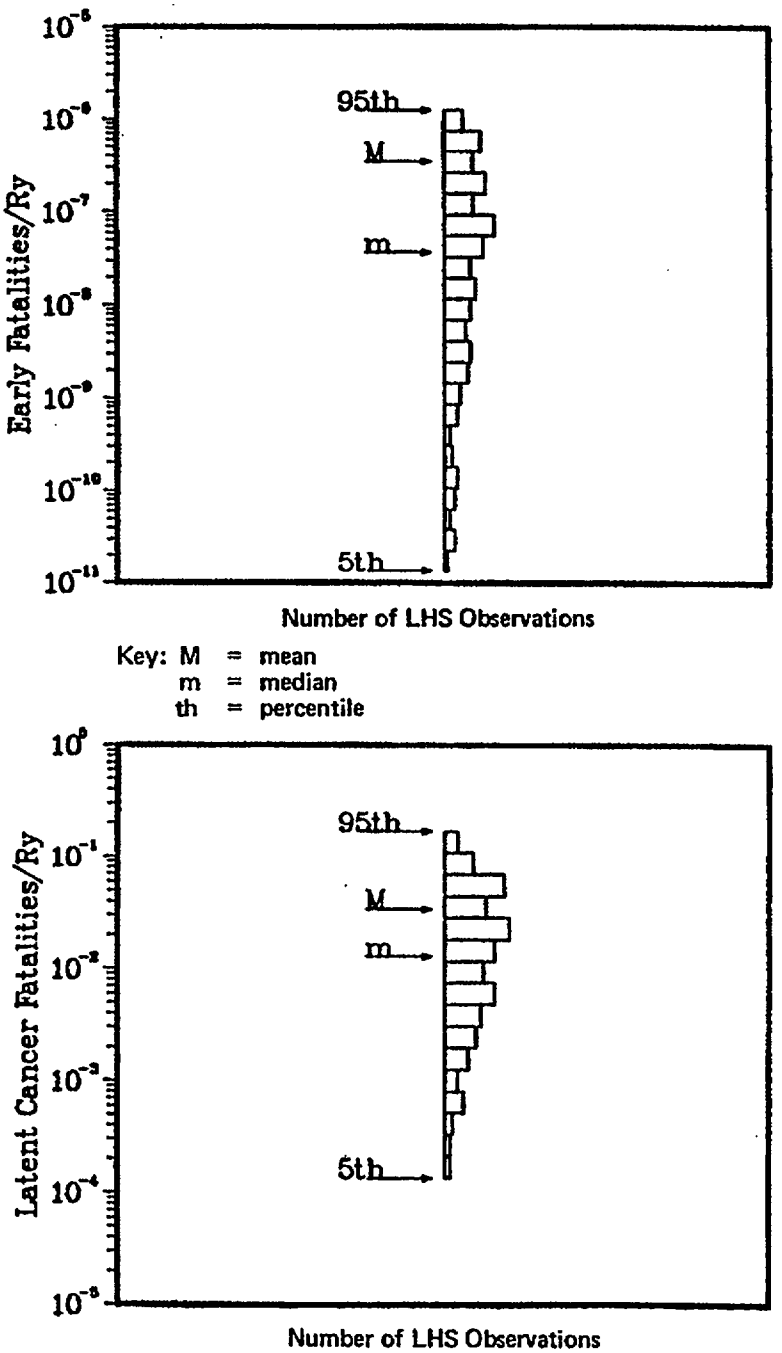
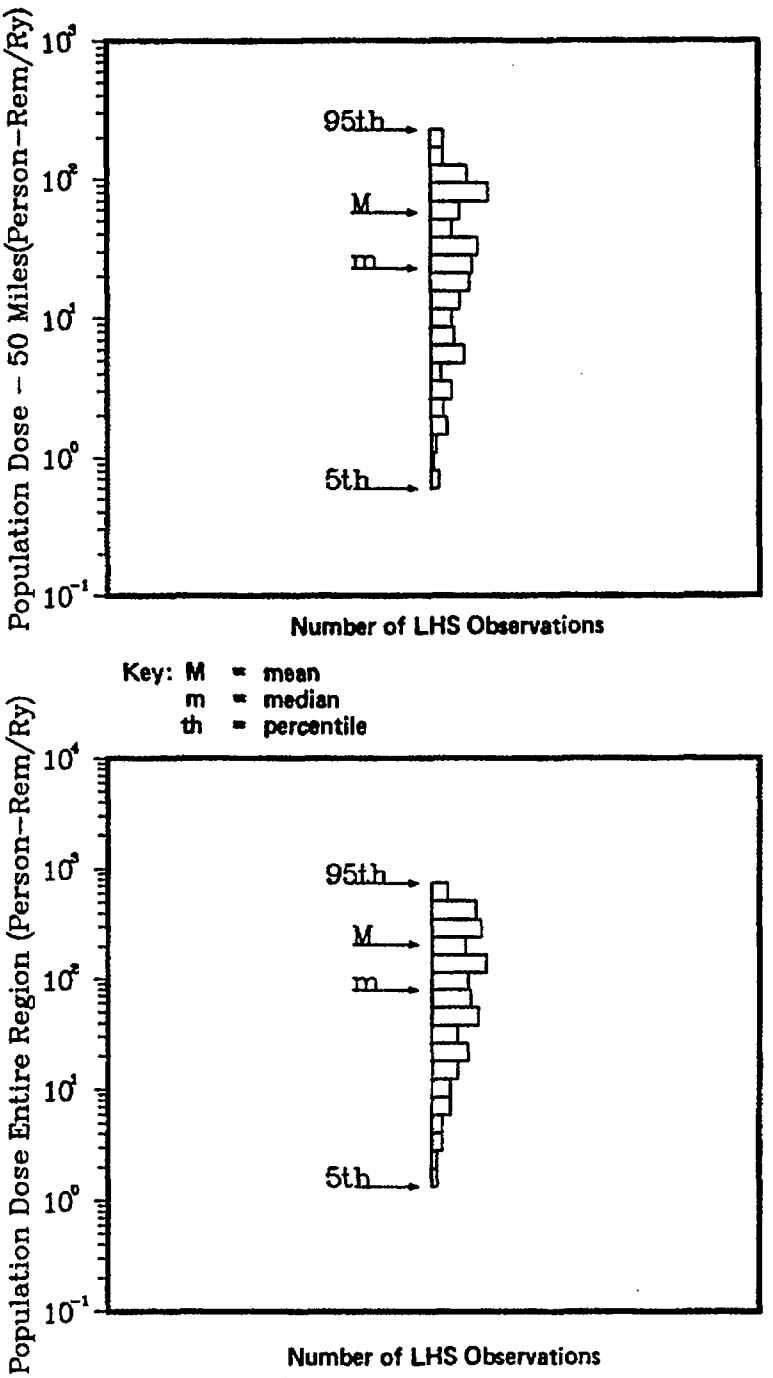


Figure 4.15 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Peach Bottom (internal initiators).



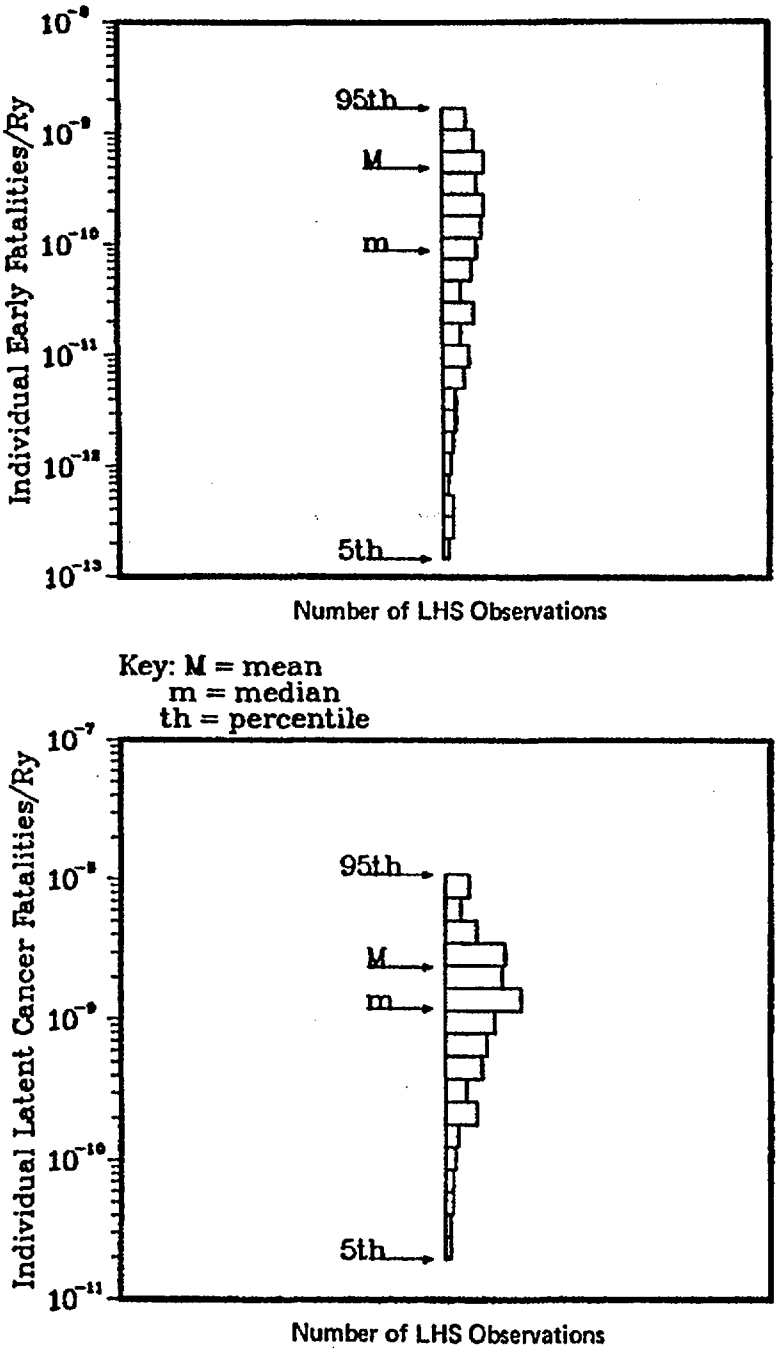
Note: As discussed in Reference 4.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

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).

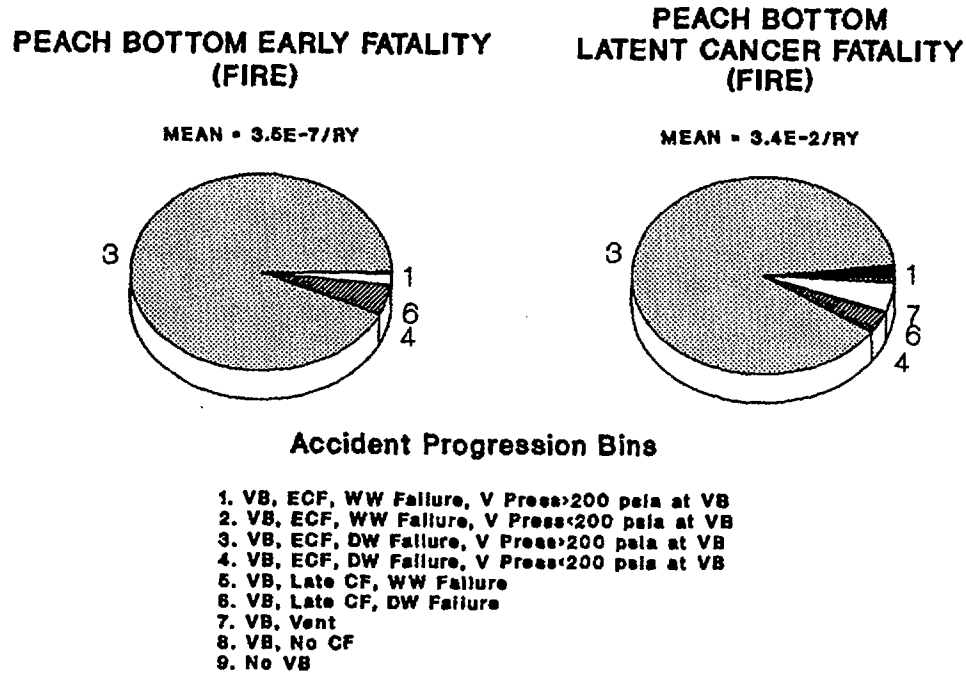


Figure 4.19 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Peach Bottom (fire initiators).

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*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

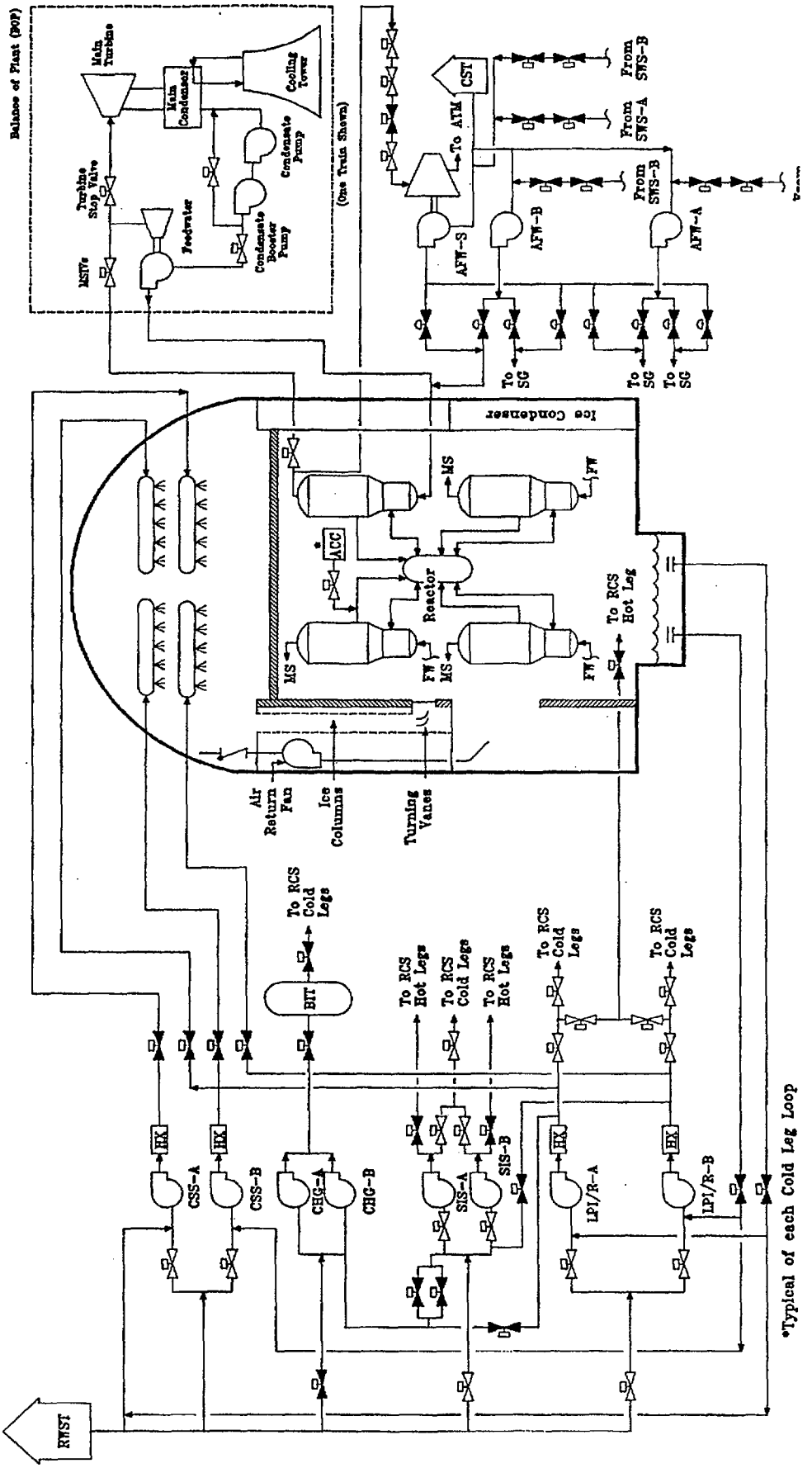
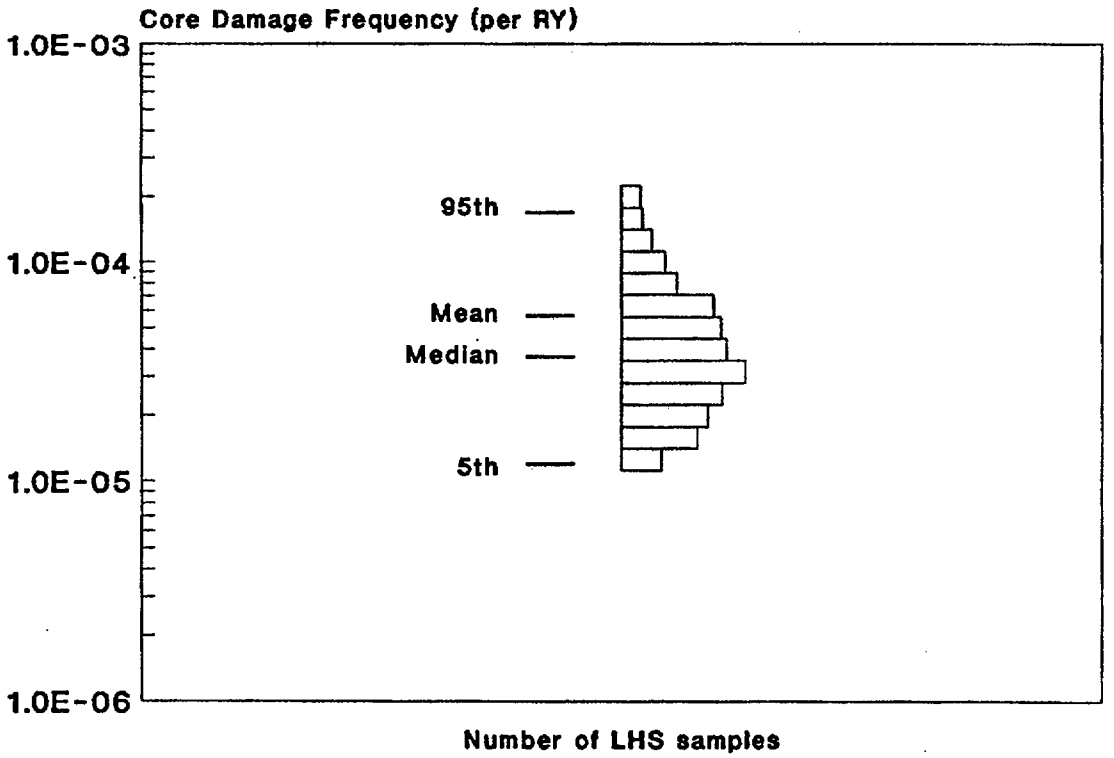


Figure 5.1 Sequoyah plant schematic.

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.

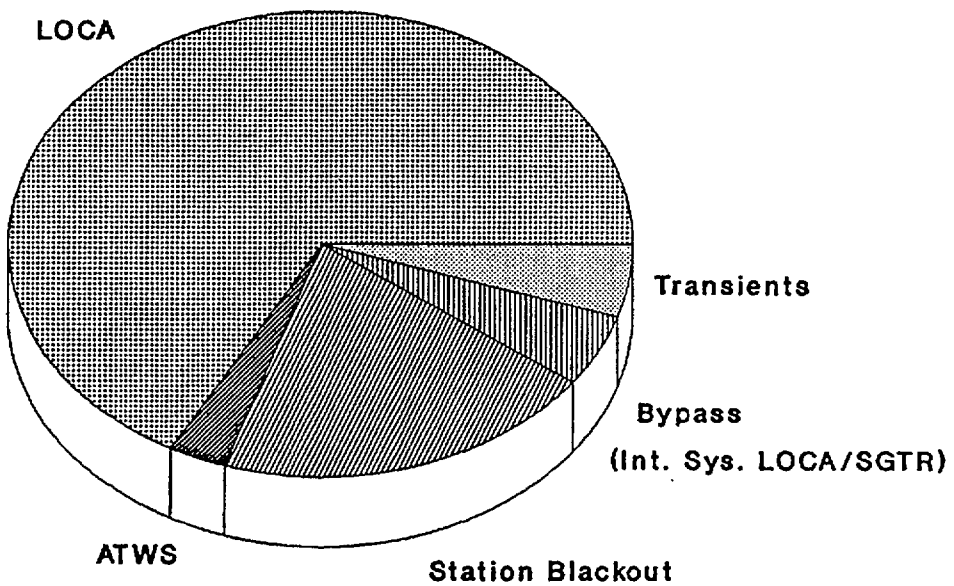


Figure 5.3 Contributors to mean core damage frequency from internal events at Sequoyah.

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.

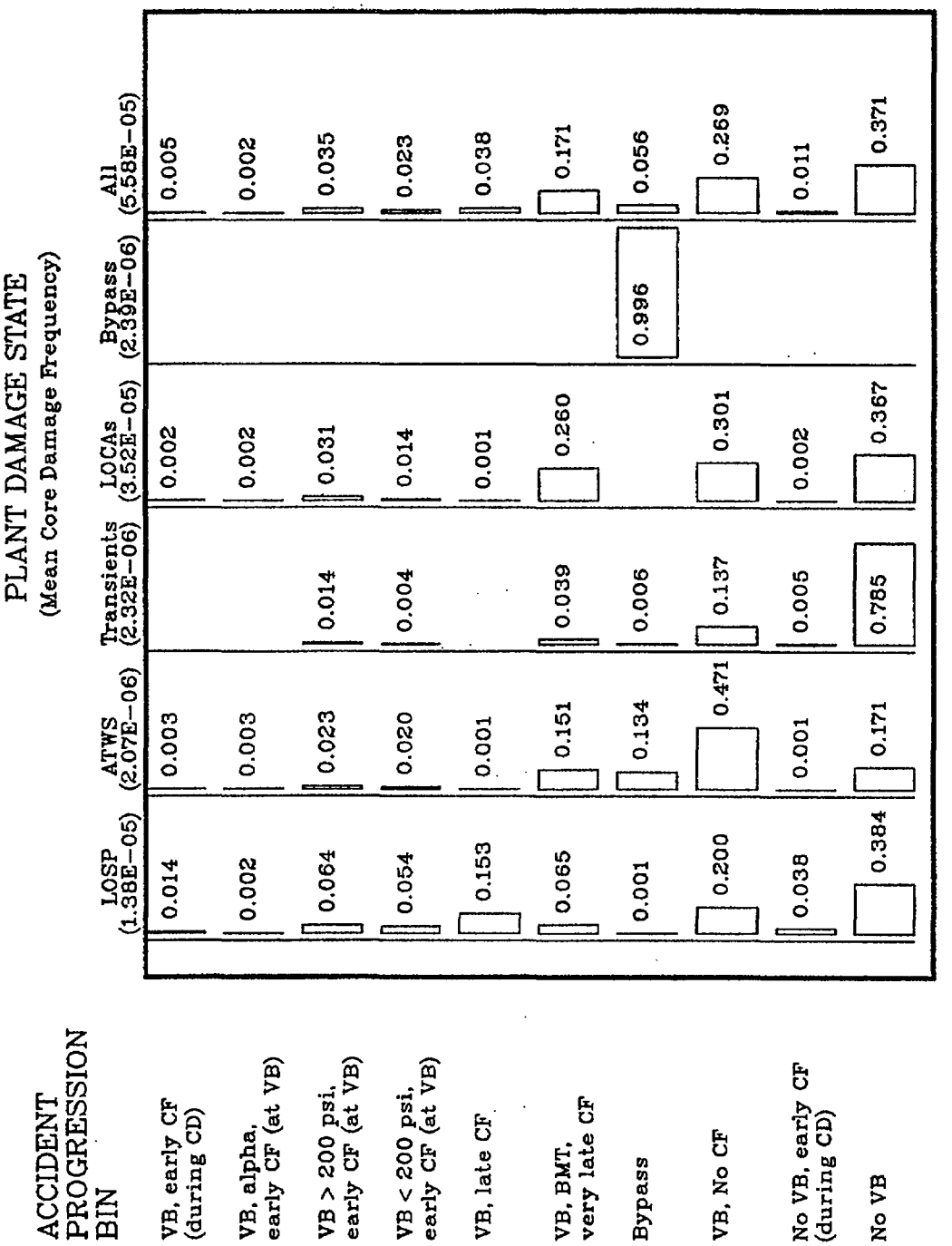


Figure 5.4 Conditional probability of accident progression bins at Sequoyah.

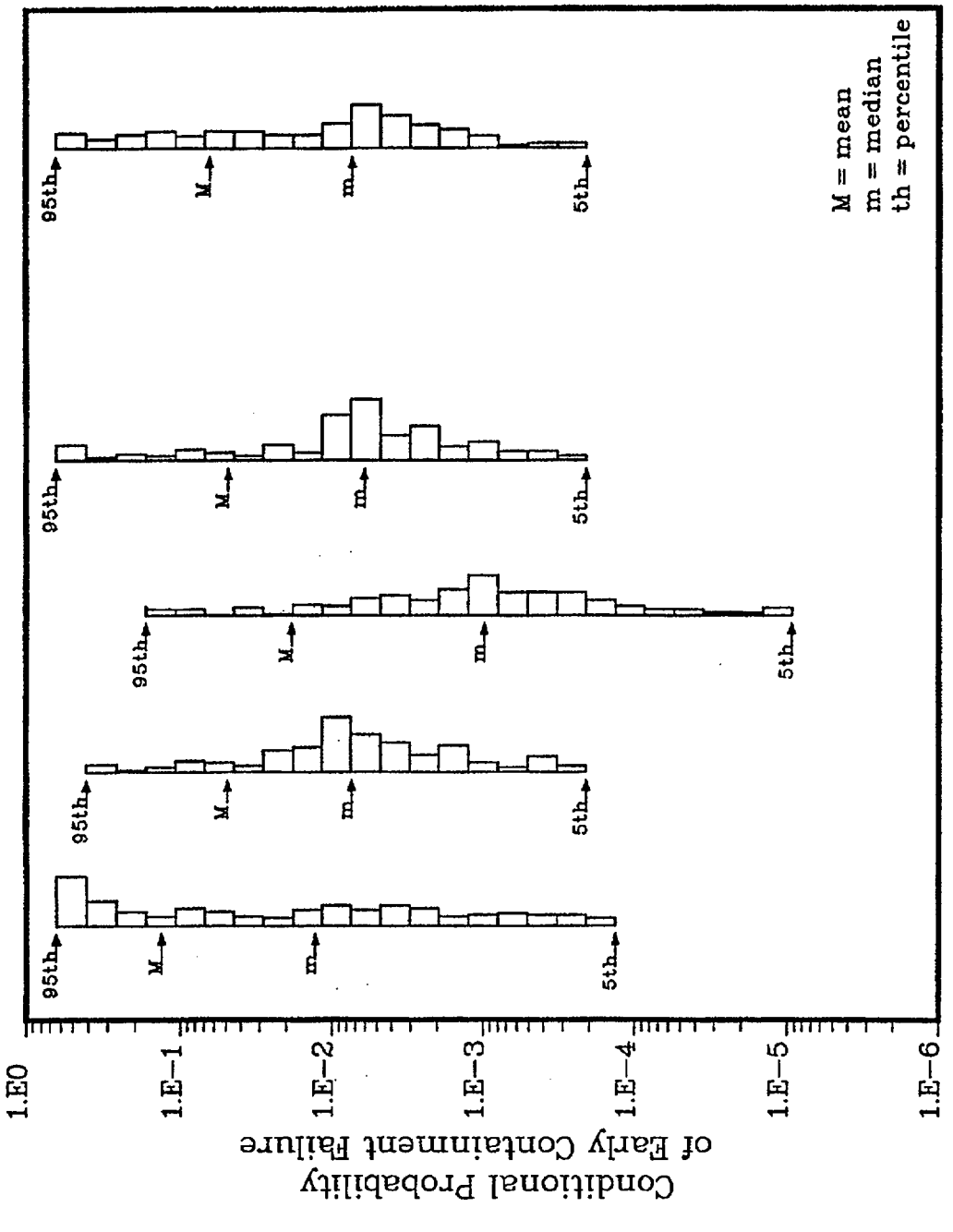


Figure 5.5 Conditional probability distributions for early containment failure at Sequoyah.

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

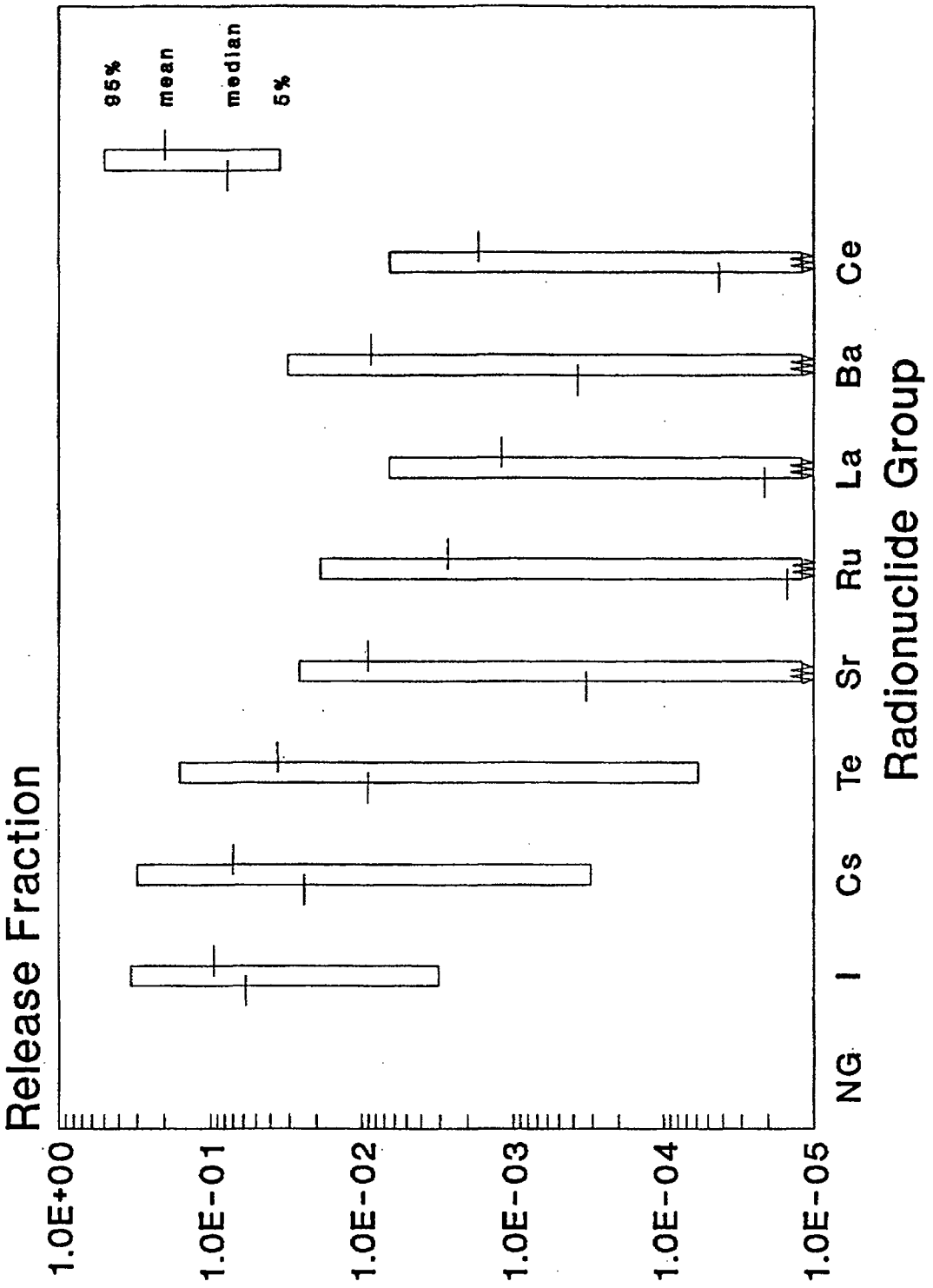


Figure 5.6 Source term distributions for early containment failure at Sequoyah.

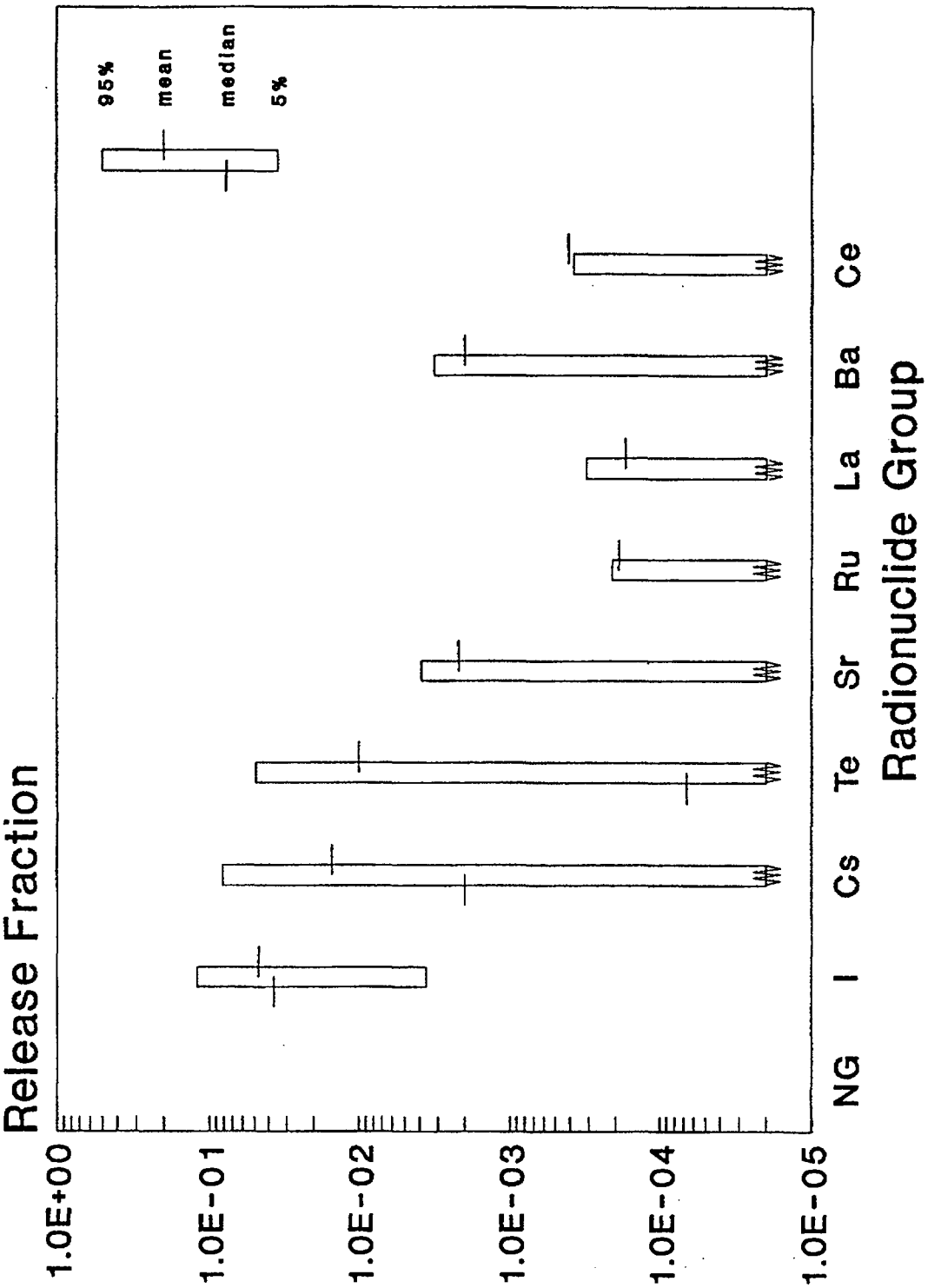


Figure 5.7 Source term distributions for late containment failure at Sequoyah.

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

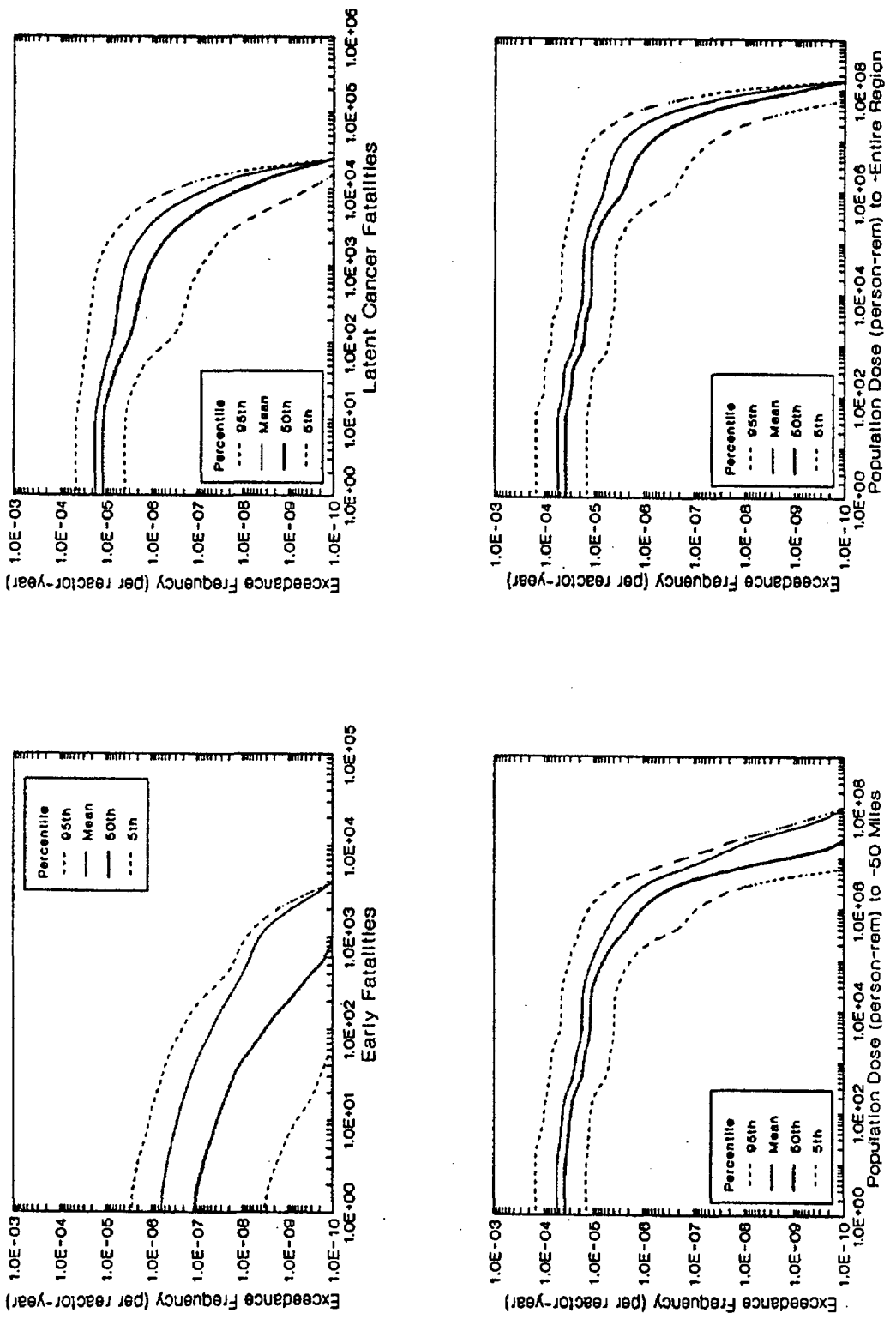
5.6.1 Results of Public Risk Estimates

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.

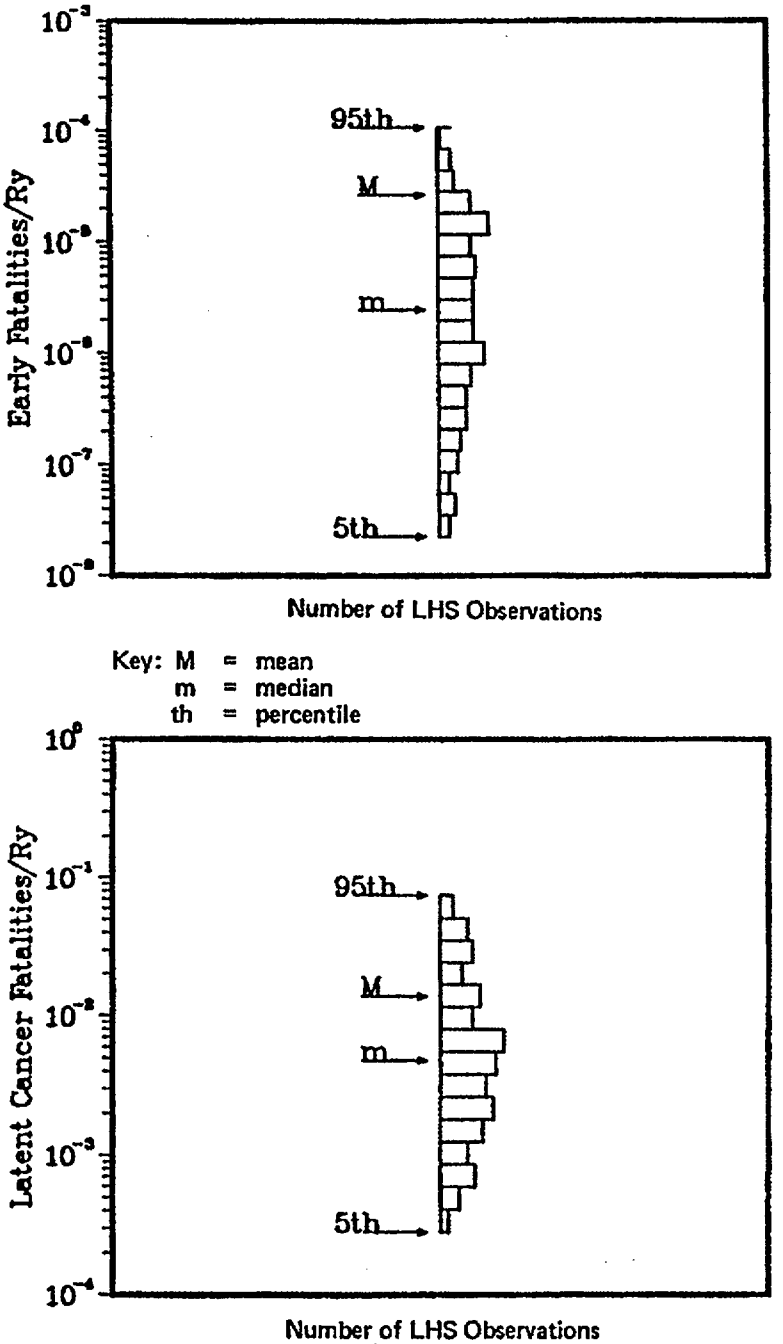
The first four of the above measures are commonly used measures in nuclear power plant risk studies. The last two are those used to compare with the NRC safety goals (Ref. 5.6).

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



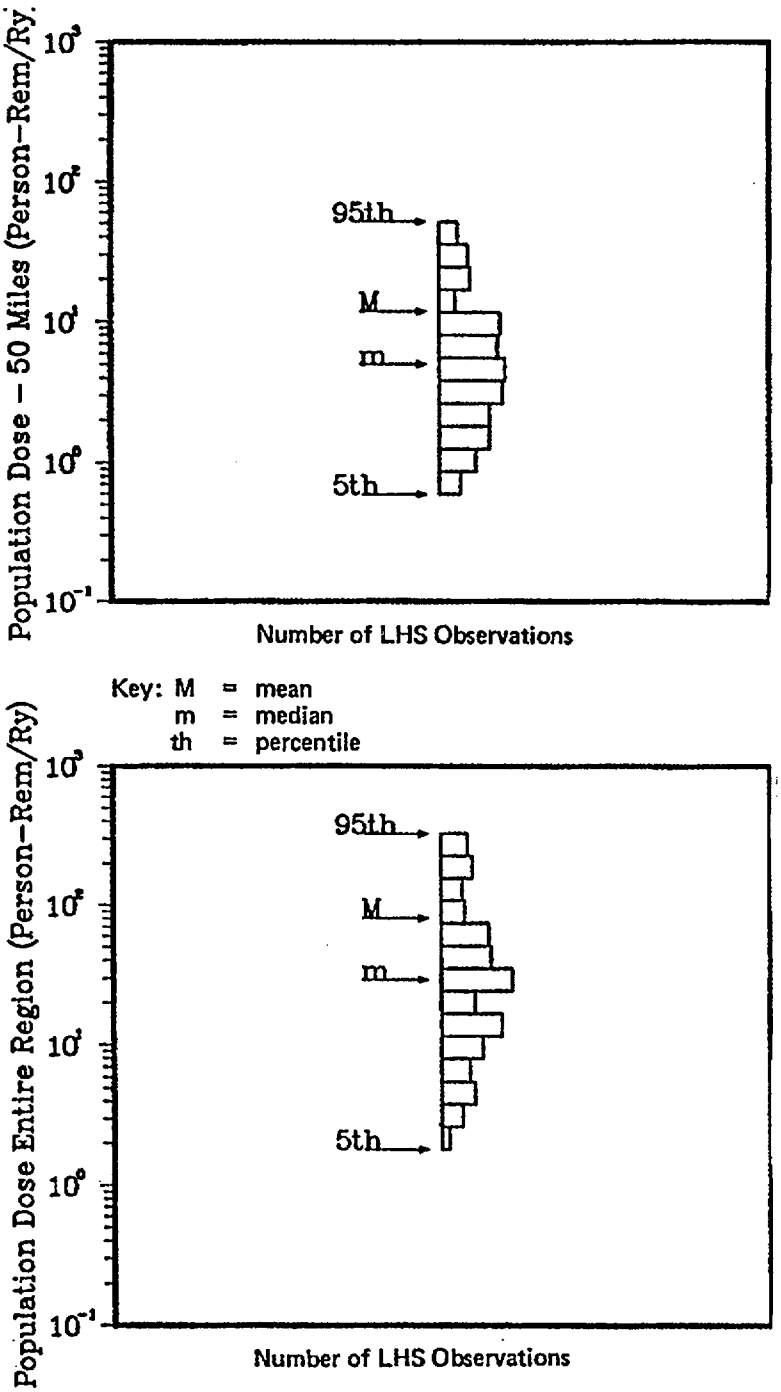
Note: As discussed in Reference 5.3, estimated consequences at frequencies at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

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



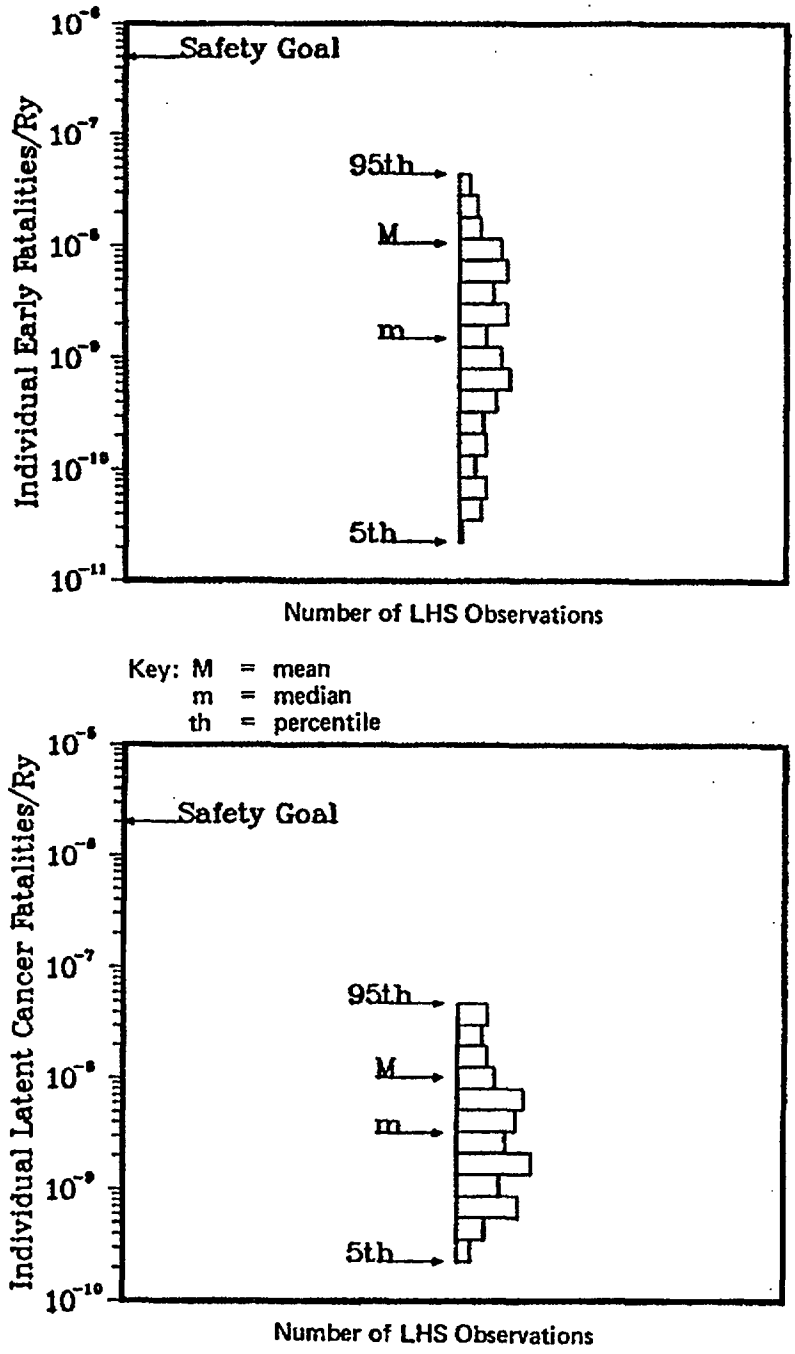
Note: As discussed in Reference 5.3, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

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.

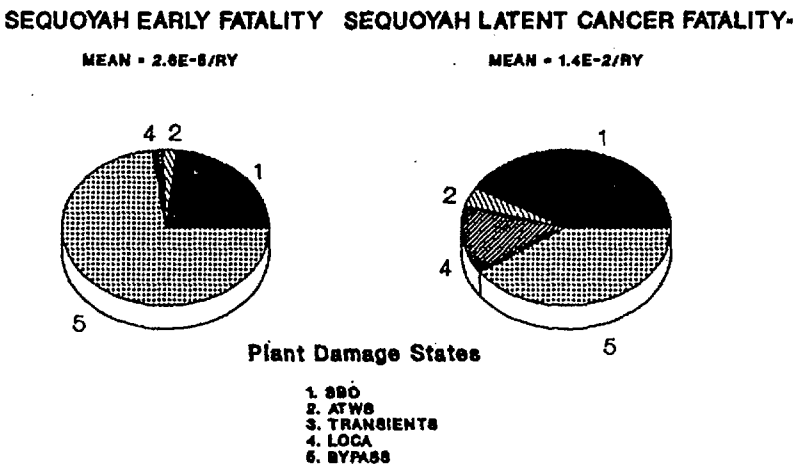


Figure 5.12 Major contributors (plant damage states) to mean early and latent cancer fatality risks at Sequoyah (internal initiators).

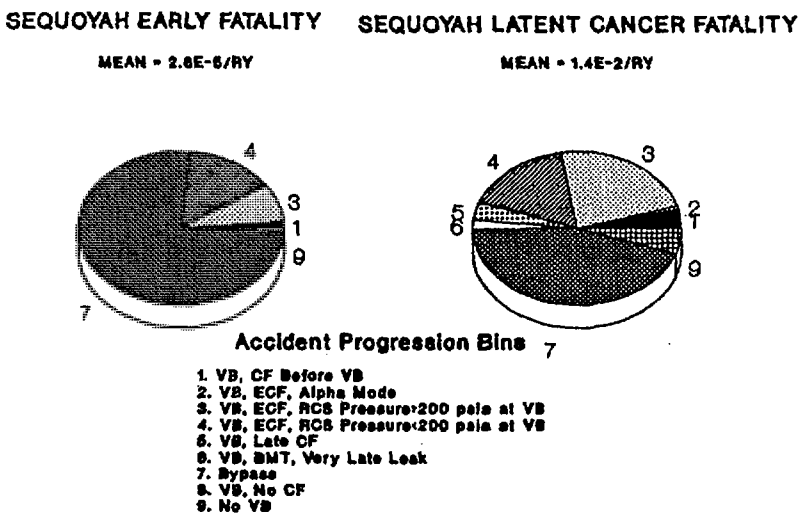


Figure 5.13 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Sequoyah (internal initiators).

REFERENCES FOR CHAPTER 5

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

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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.

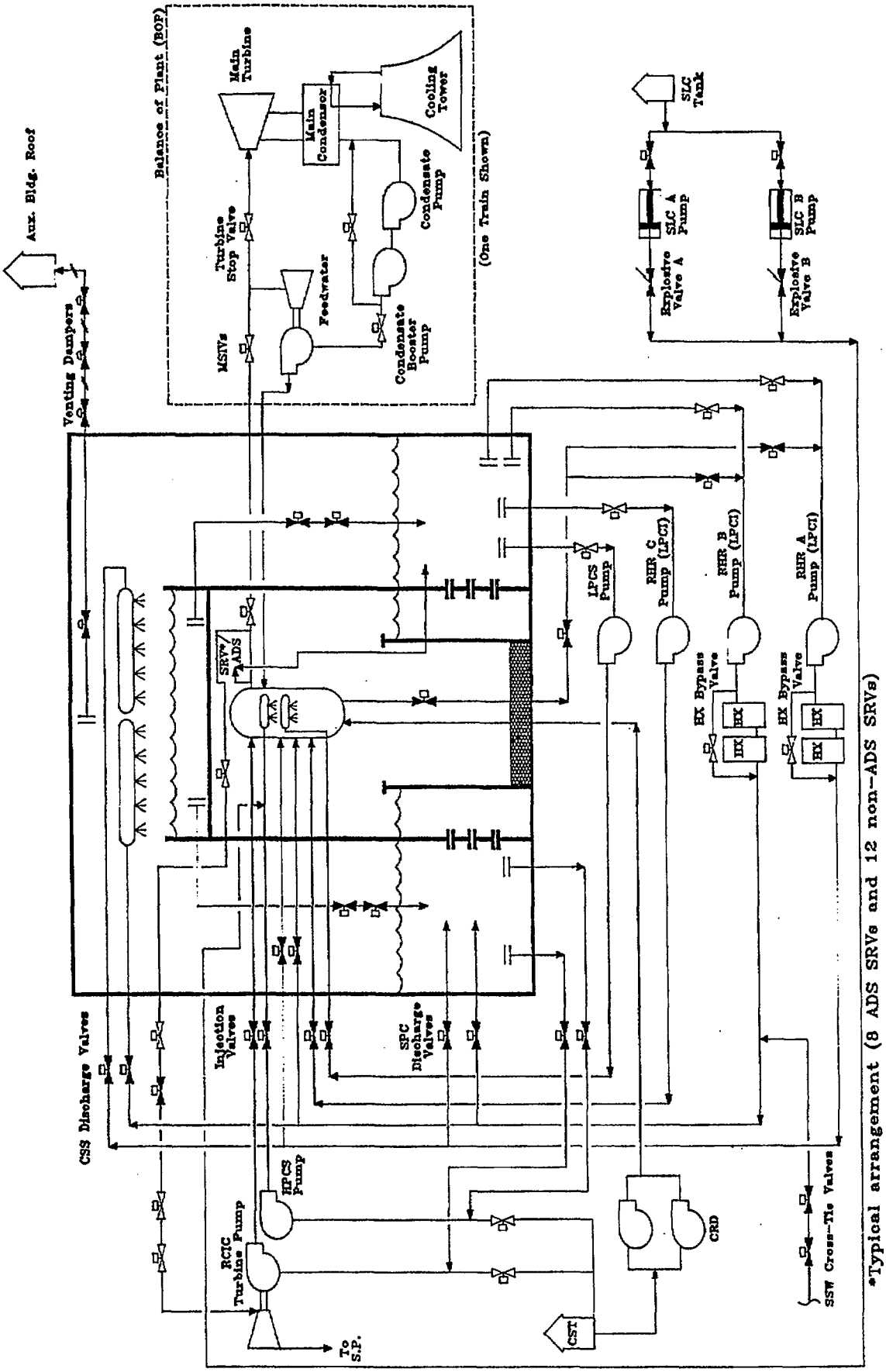
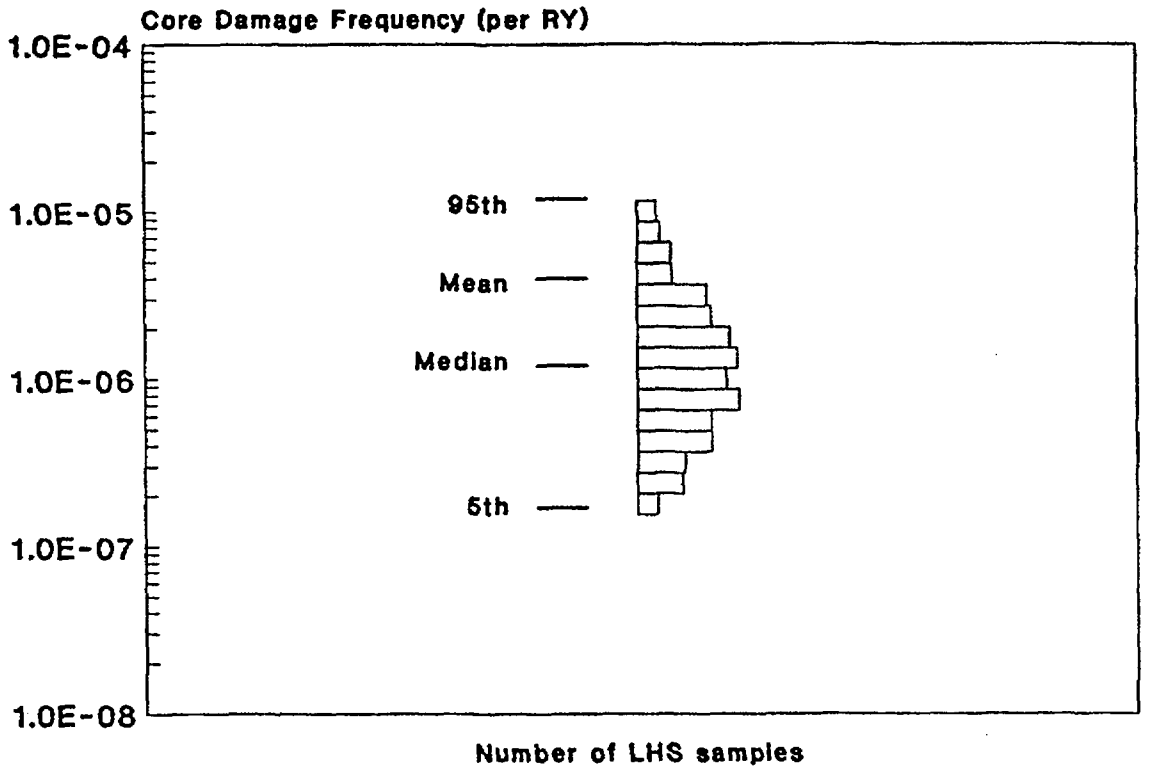


Figure 6.1 Grand Gulf plant schematic.

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.

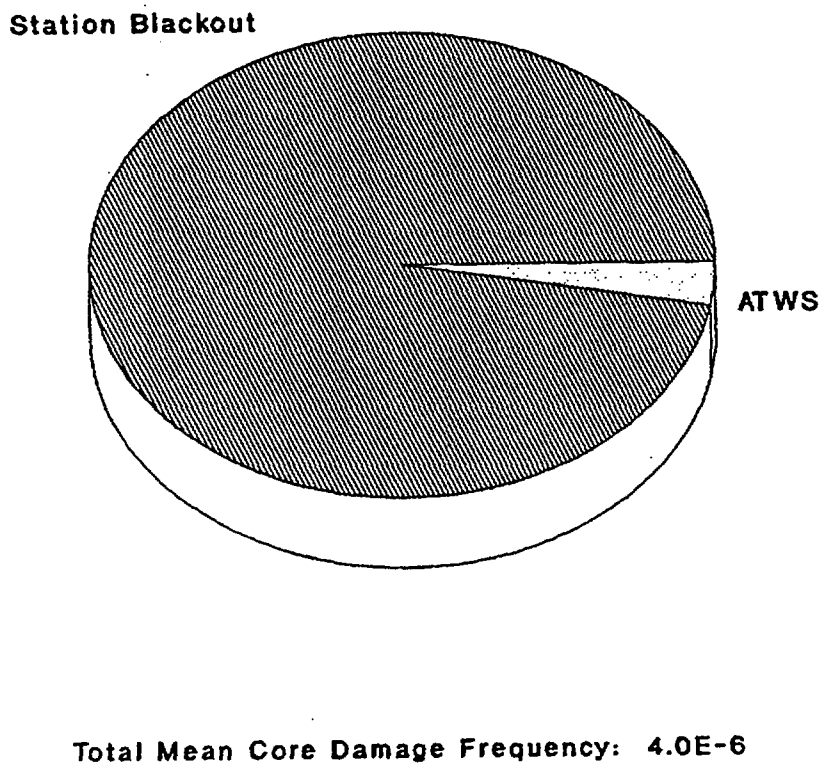


Figure 6.3 Contributors to mean core damage frequency from internal events at Grand Gulf.

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.

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

*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.

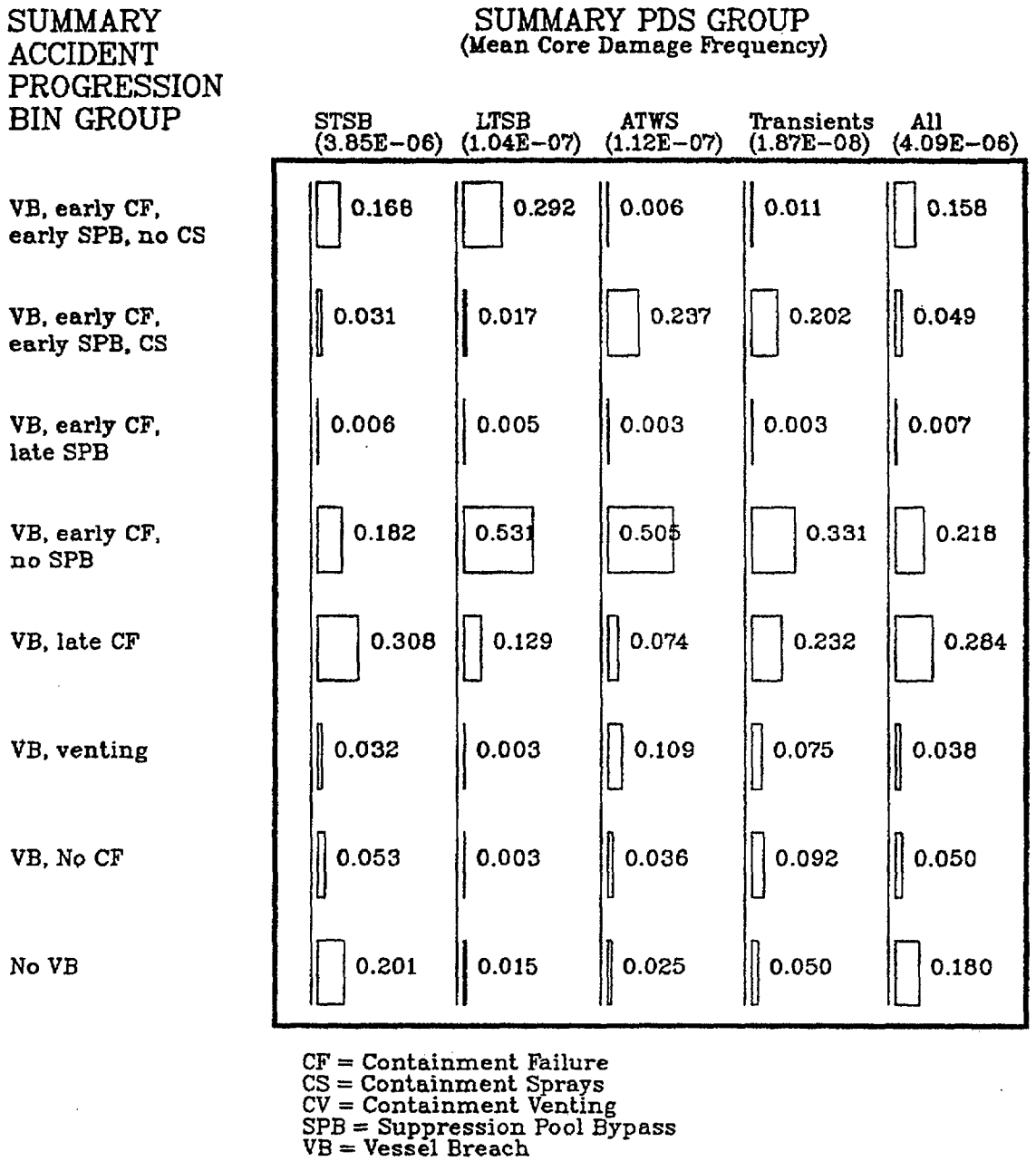


Figure 6.4 Conditional probability of accident progression bins at Grand Gulf.

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.

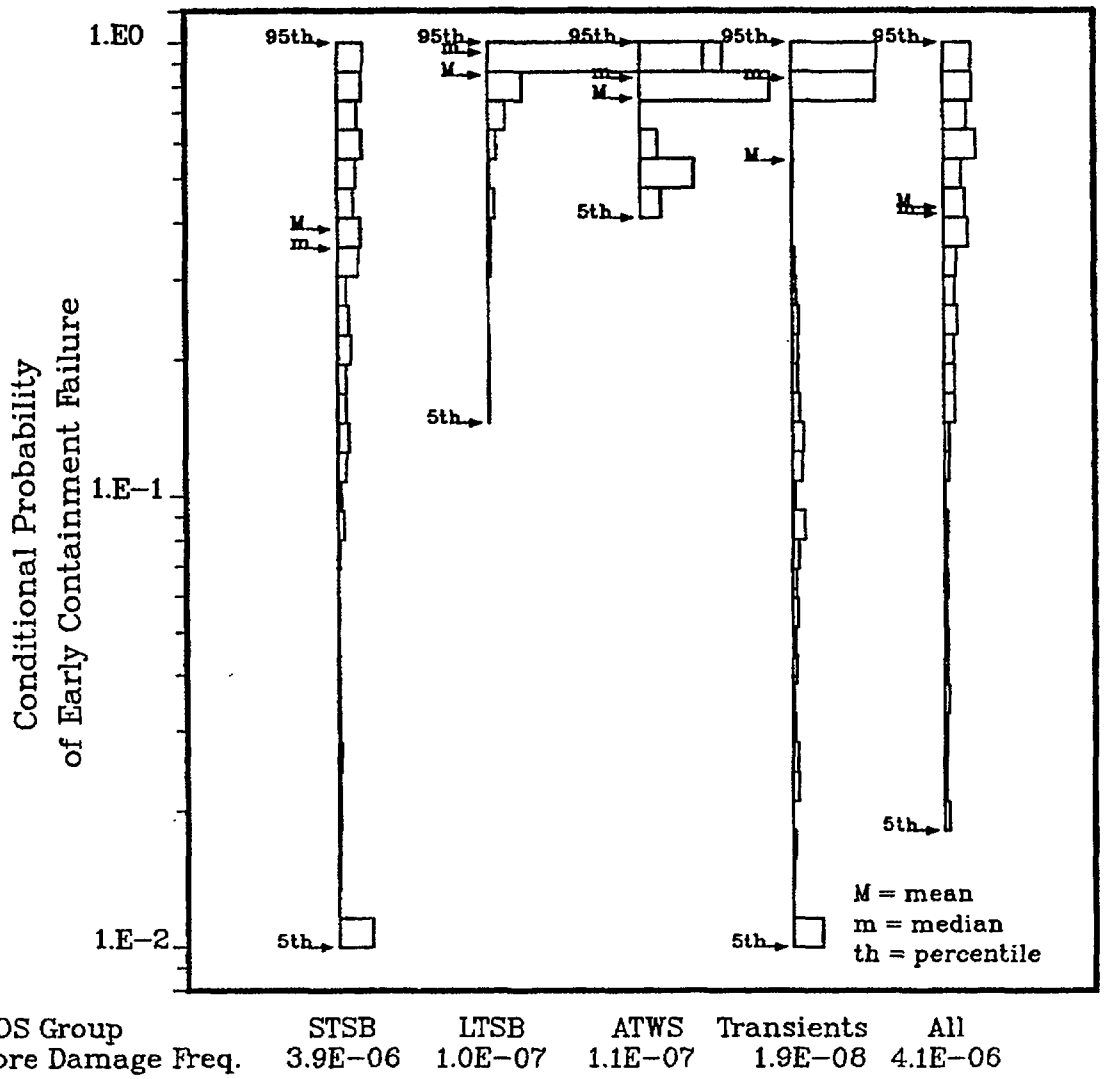


Figure 6.5 Conditional probability distributions for early containment failure at Grand Gulf.

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,

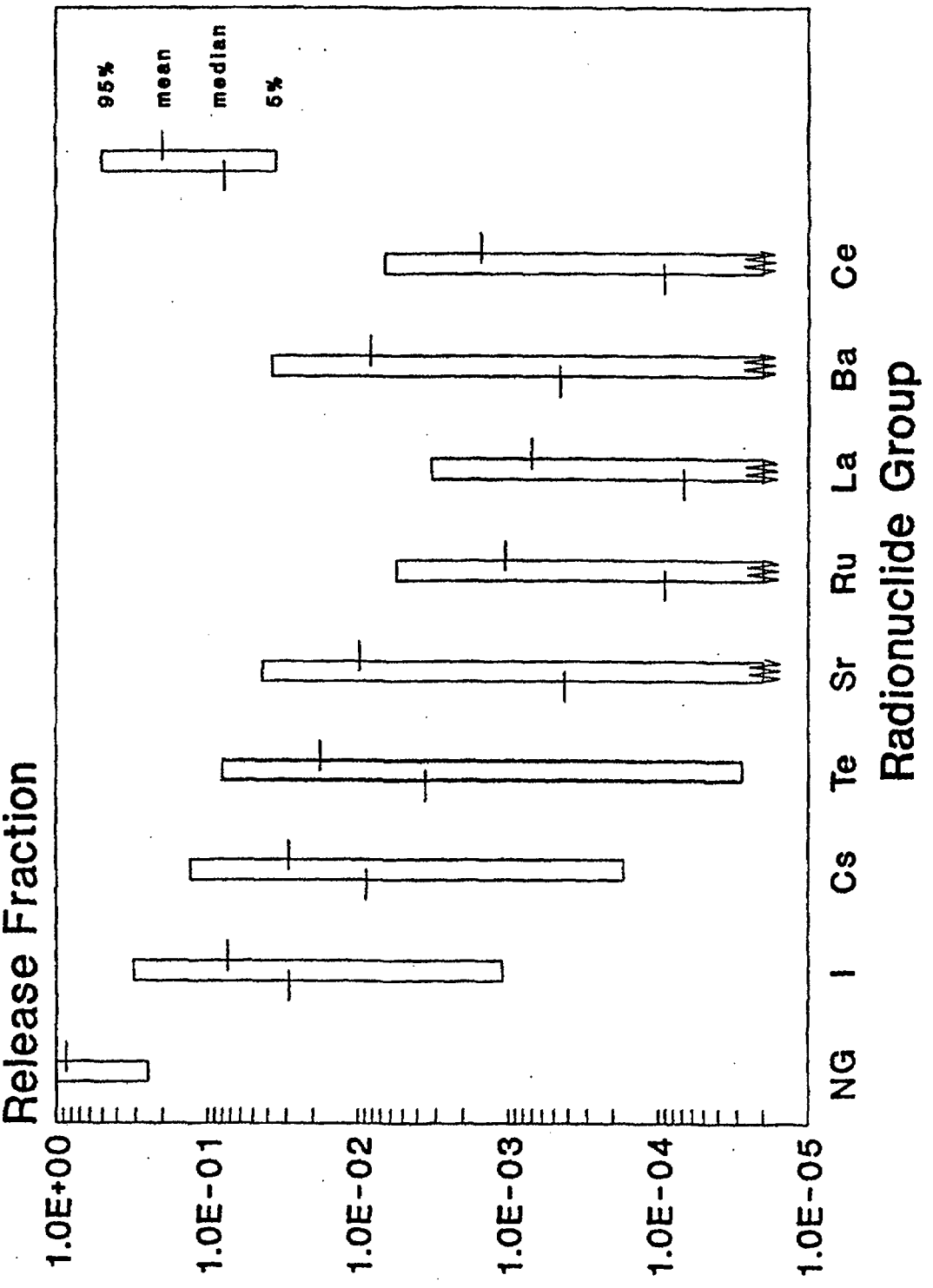


Figure 6.6 Source term distributions for early containment failure with drywell failed and sprays unavailable at Grand Gulf.

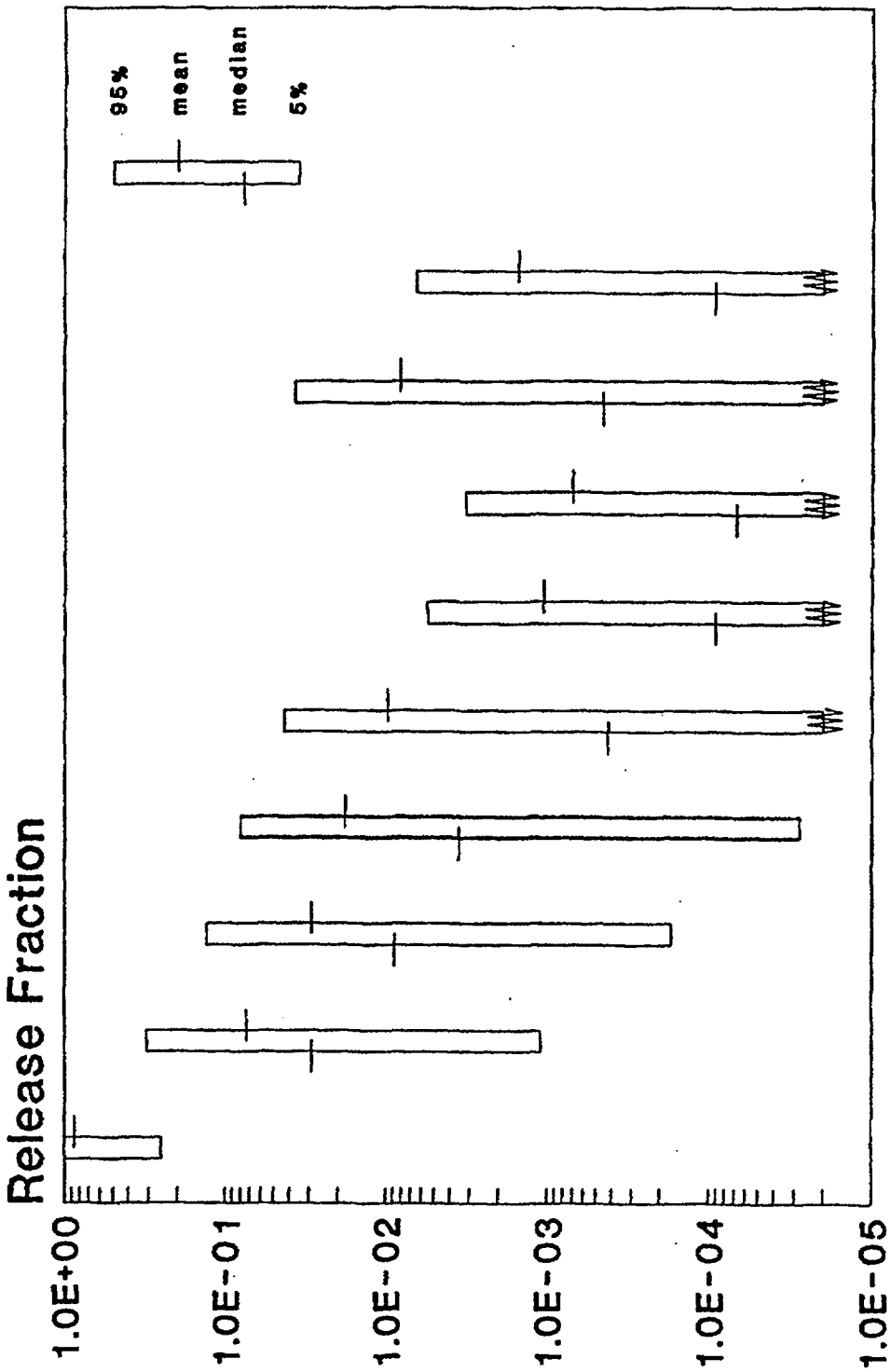
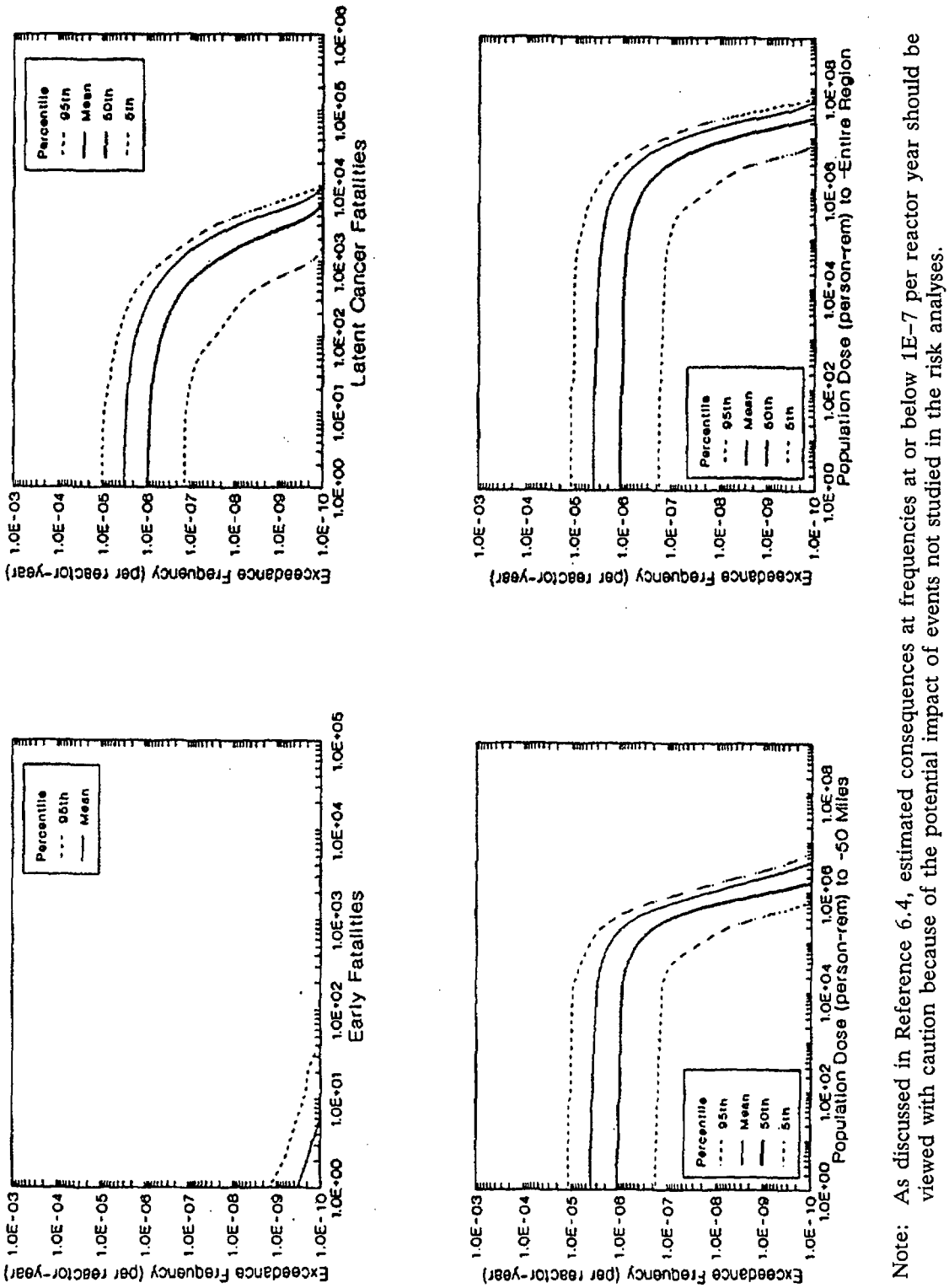


Figure 6.7 Source term distributions for early containment failure with drywell intact at Grand Gulf.



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.

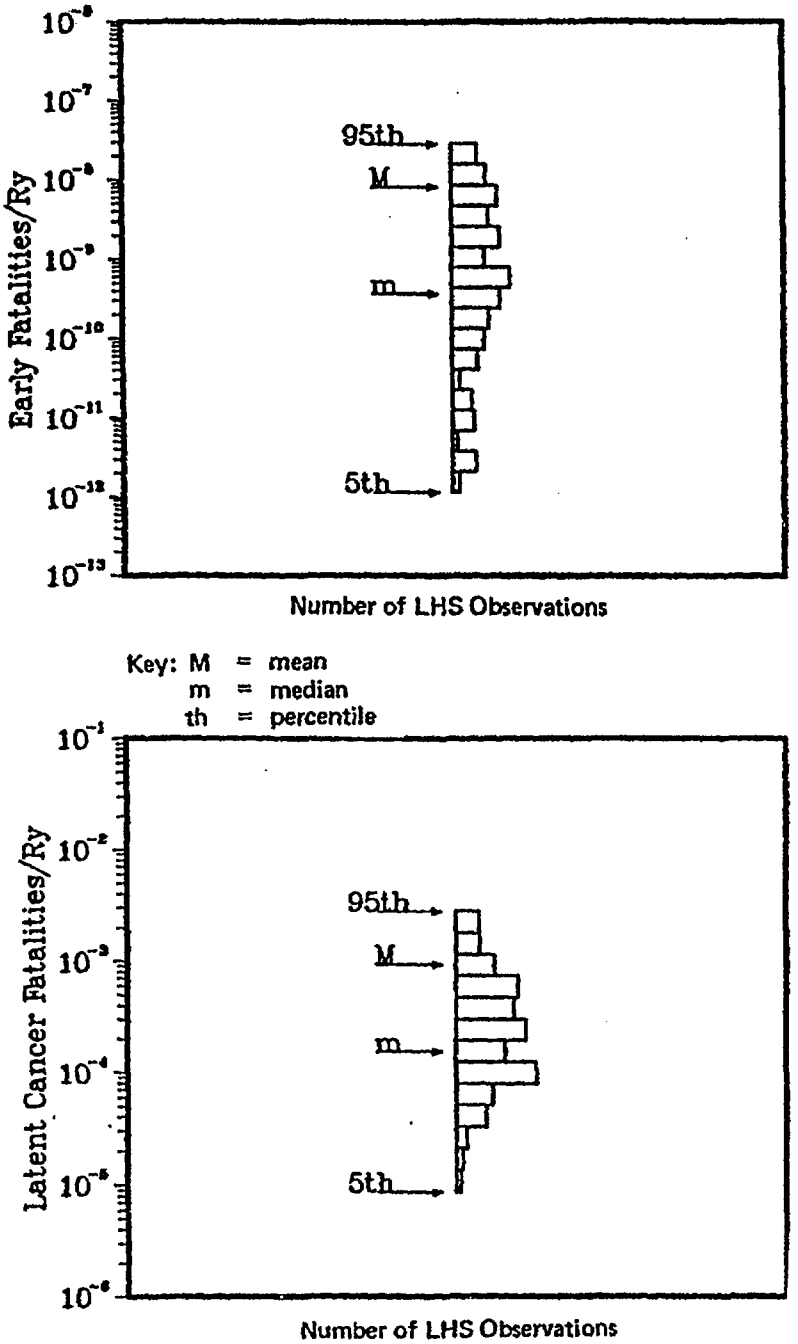
The results in Figure 6.9 have been analyzed to identify the relative contributions of accident sequences and containment failure modes to mean risk. These results are presented in Figures 6.12 and 6.13. As may be seen, the mean early fatality risk at Grand Gulf is dominated by short-term station blackout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

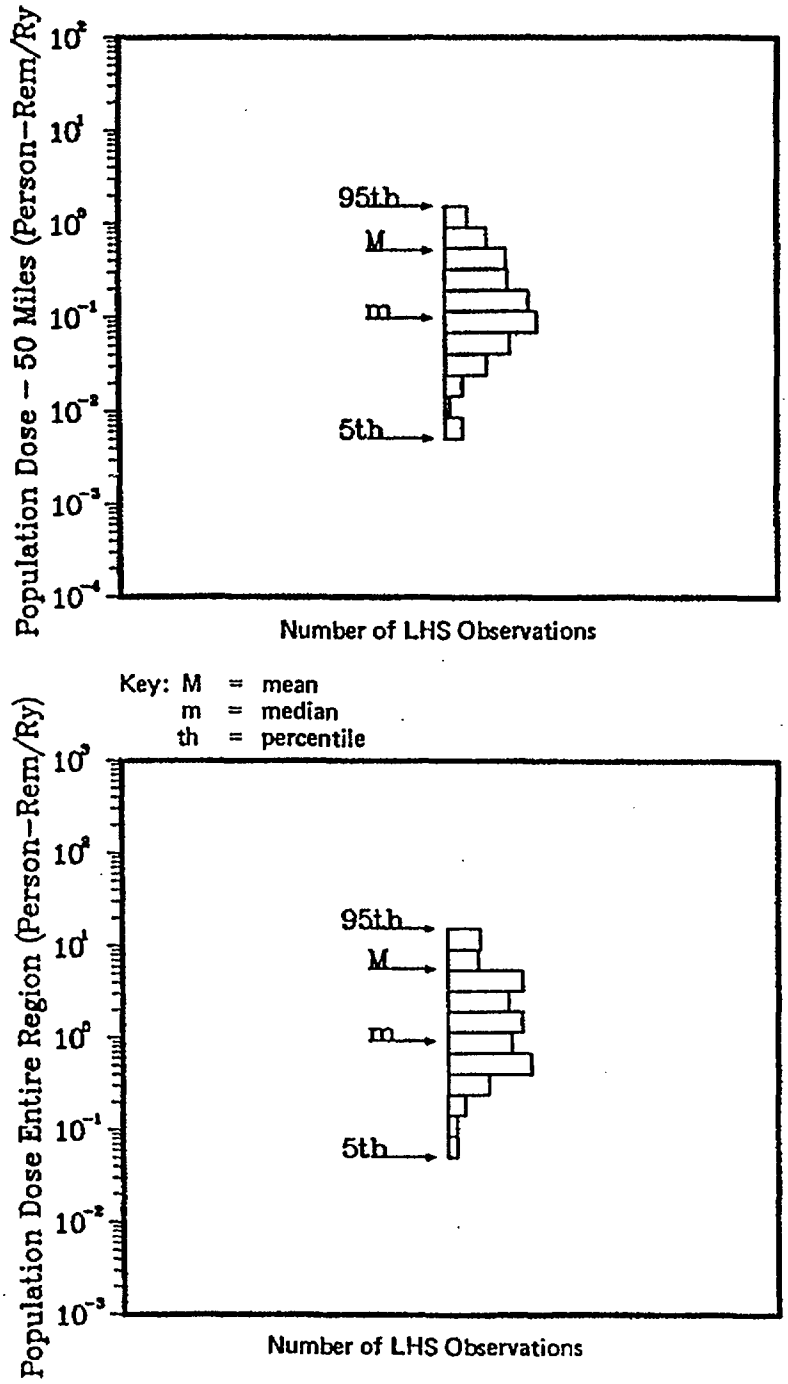
As mentioned before, risk to the public from the operation of the Grand Gulf plant is lower than the other four plants in this study. Some of the plant features that contribute to these low risk estimates are described below.

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



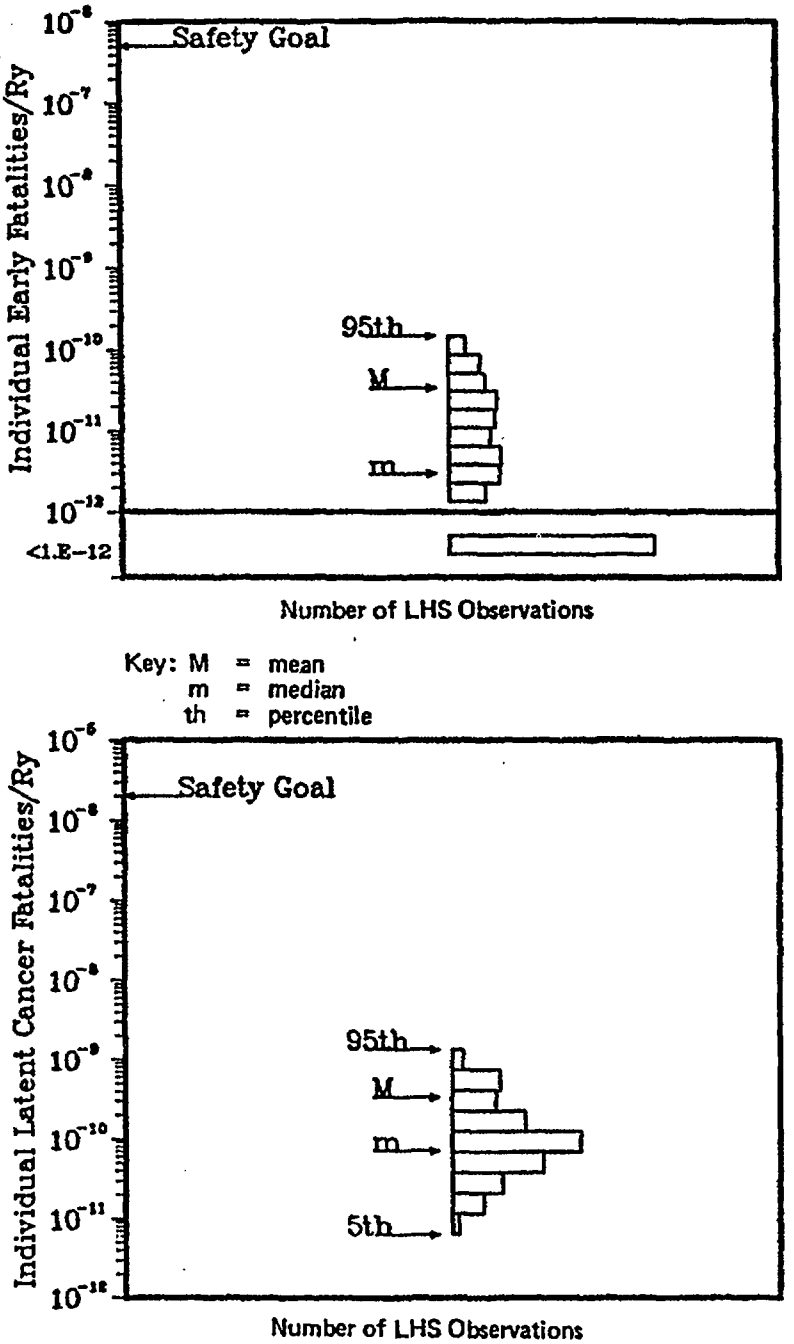
Note: As discussed in Reference 6.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

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).

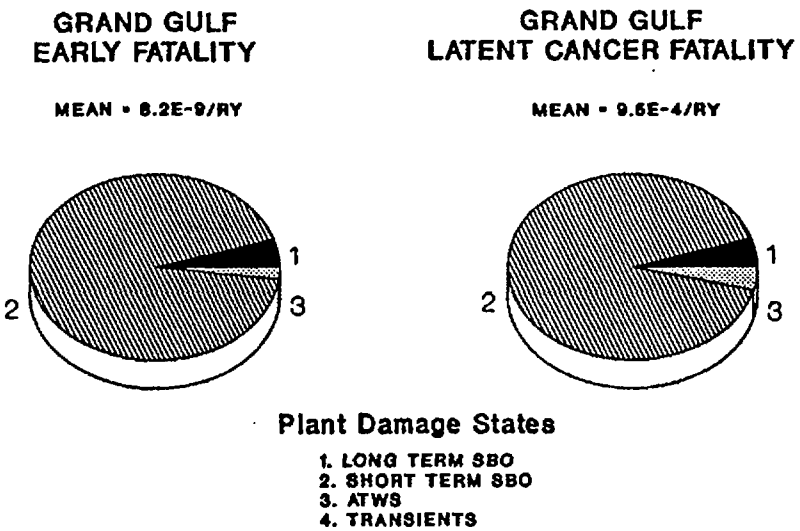


Figure 6.12 Major contributors (plant damage states) to mean early and latent cancer fatality risks at Grand Gulf (internal initiators).

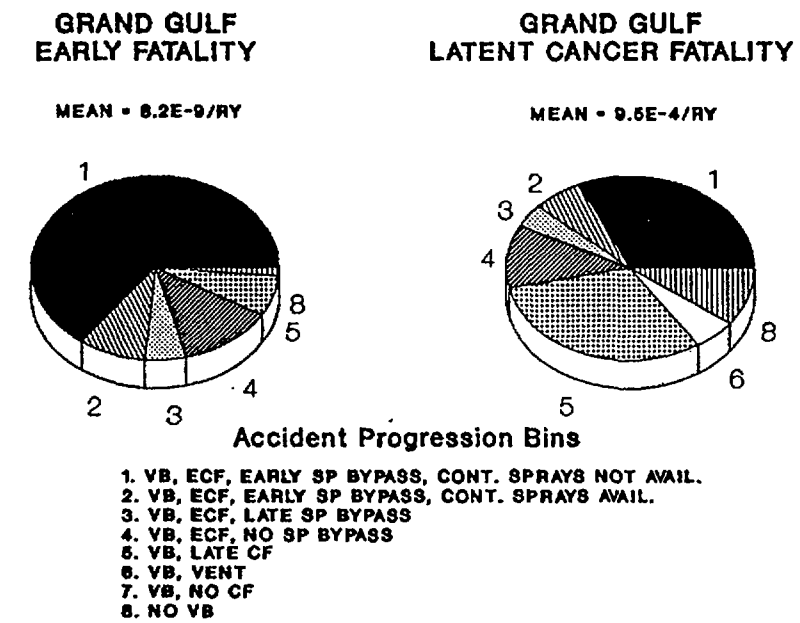


Figure 6.13 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Grand Gulf (internal initiators).

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

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

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

6.3 S. W. Hatch et al., "Reactor Safety Study Methodology Applications Program: Grand Gulf No. 1 BWR Power Plant," Sandia National Laboratories and Battelle Columbus Laboratories, NUREG/CR-1659/4 of 4, SAND80-1897/4 of 4, November 1981.

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

6.5 A. D. Swain III, "Accident Sequence Evaluation Program—Human Reliability Analysis Procedure," Sandia National Laboratories, NUREG/CR-4772, SAND86-1996, February 1987.

6.6 Mississippi Power & Light Company, "Evacuation Time Estimates for the Grand Gulf Nuclear Station Plume Exposure Pathway Emergency Planning Zone," Revision 4, March 1986.

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

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

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

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.

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),

*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).

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.

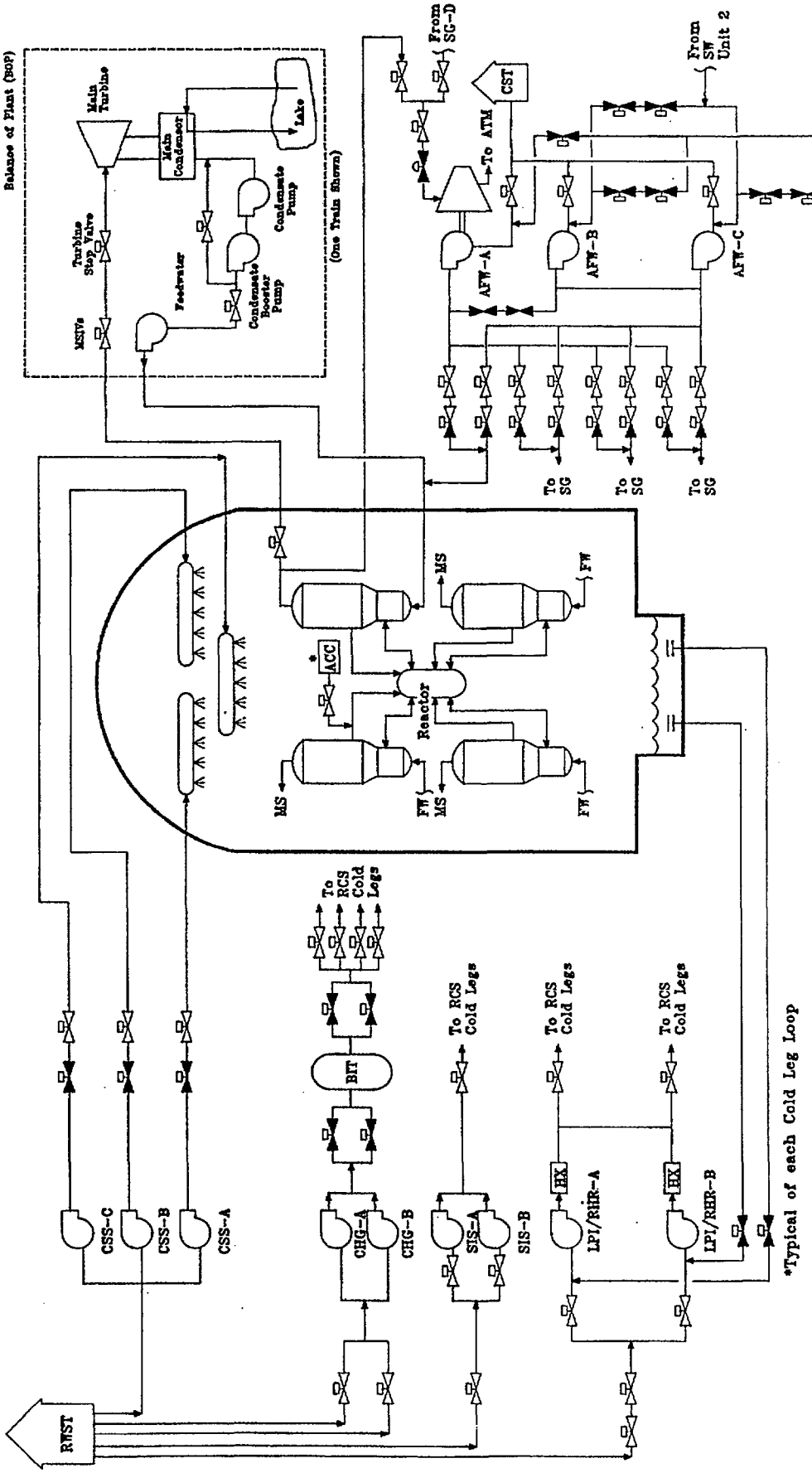


Figure 7.1 Zion plant schematic.

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)

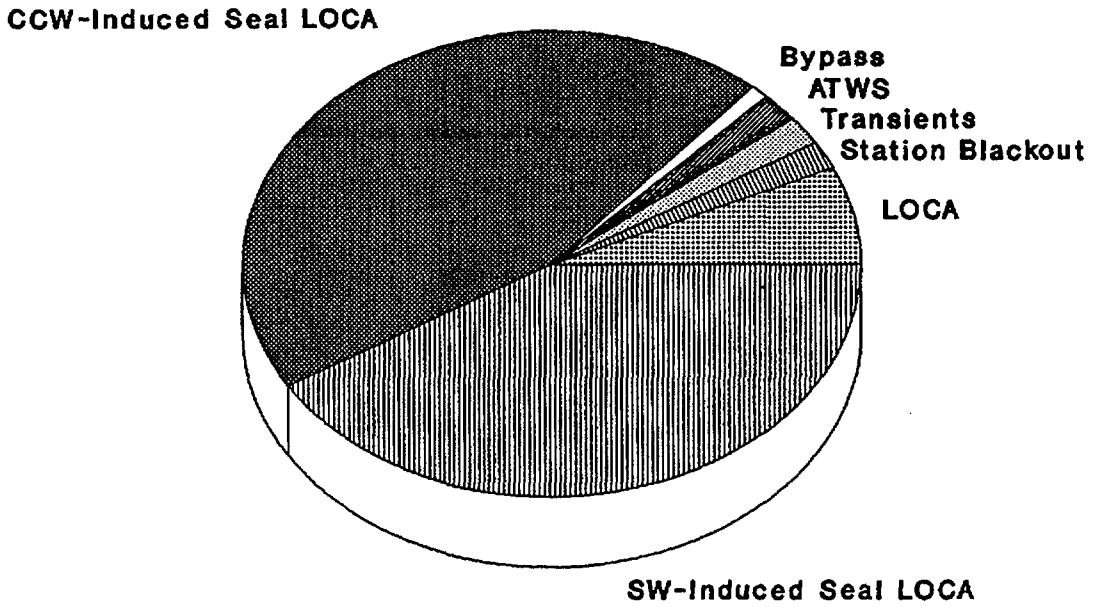
Characteristics of the Zion plant design and operation that have been found to be important in the analysis of the core damage frequency include:

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.



Total Mean Core Damage Frequency: 3.4E-4

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.

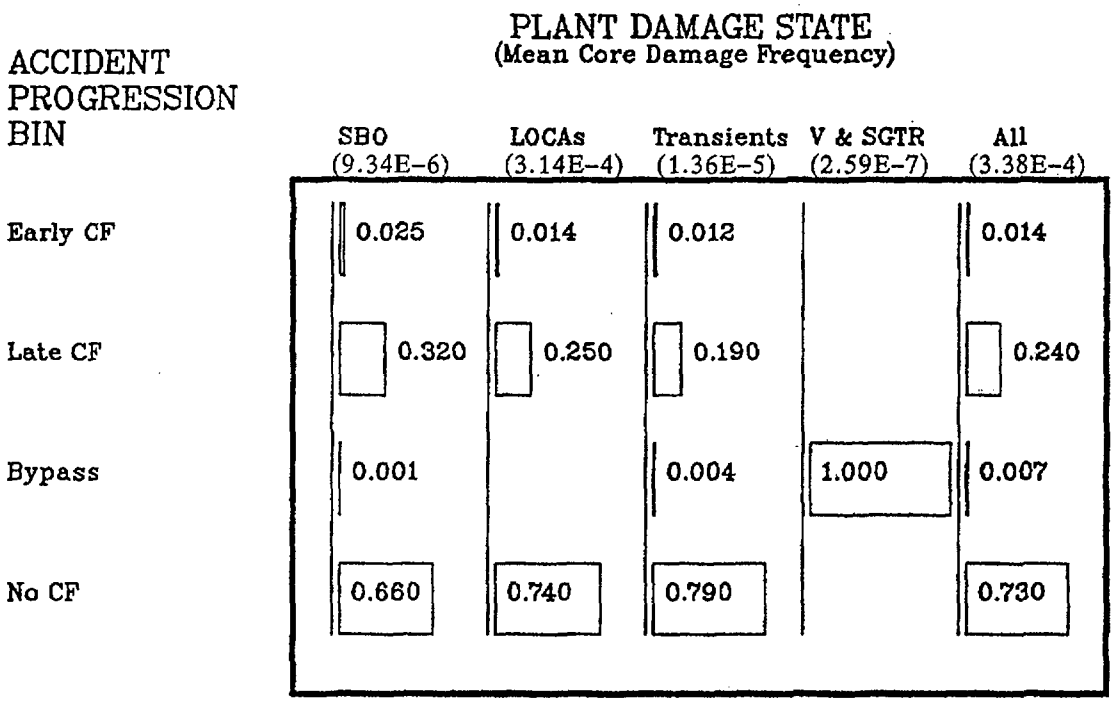
Switchover to recirculation cooling and initiation of feed and bleed cooling were included in the original Zion Probabilistic Safety Study and have been given close scrutiny by the licensee. Each one of these actions is present in the emergency procedures. Appropriate consideration of the procedures, 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

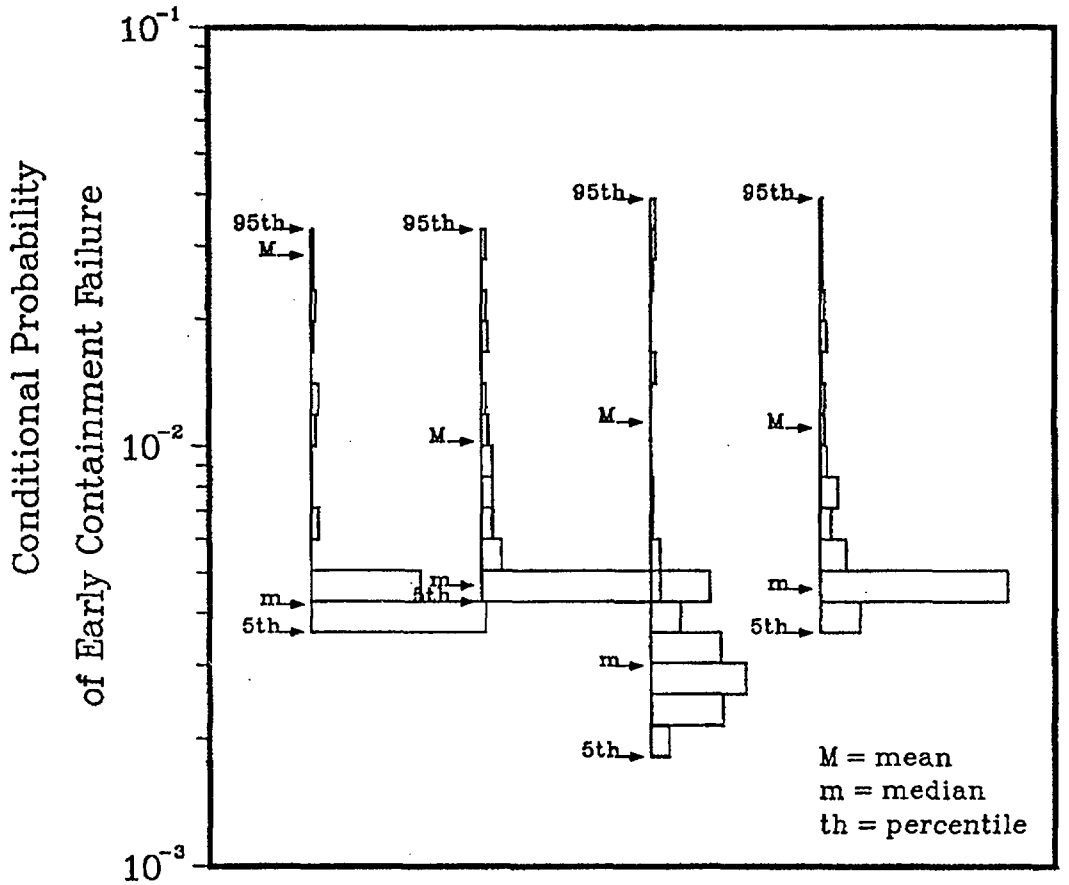
The Zion containment consists of a large, dry containment building that houses the reactor pressure vessel, reactor coolant system piping, and the secondary system's steam generators. The containment building is a prestressed concrete structure with a steel liner. This building has a volume of 2.6 million 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



Key: CF = Containment Failure

Figure 7.3 Conditional probability of accident progression bins at Zion.



Plant Damage States	SBO	LOCAs	Transients	All
Core Damage Freq.	(9.34E-6)	(3.14E-4)	(1.36E-5)	(3.38E-4)

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

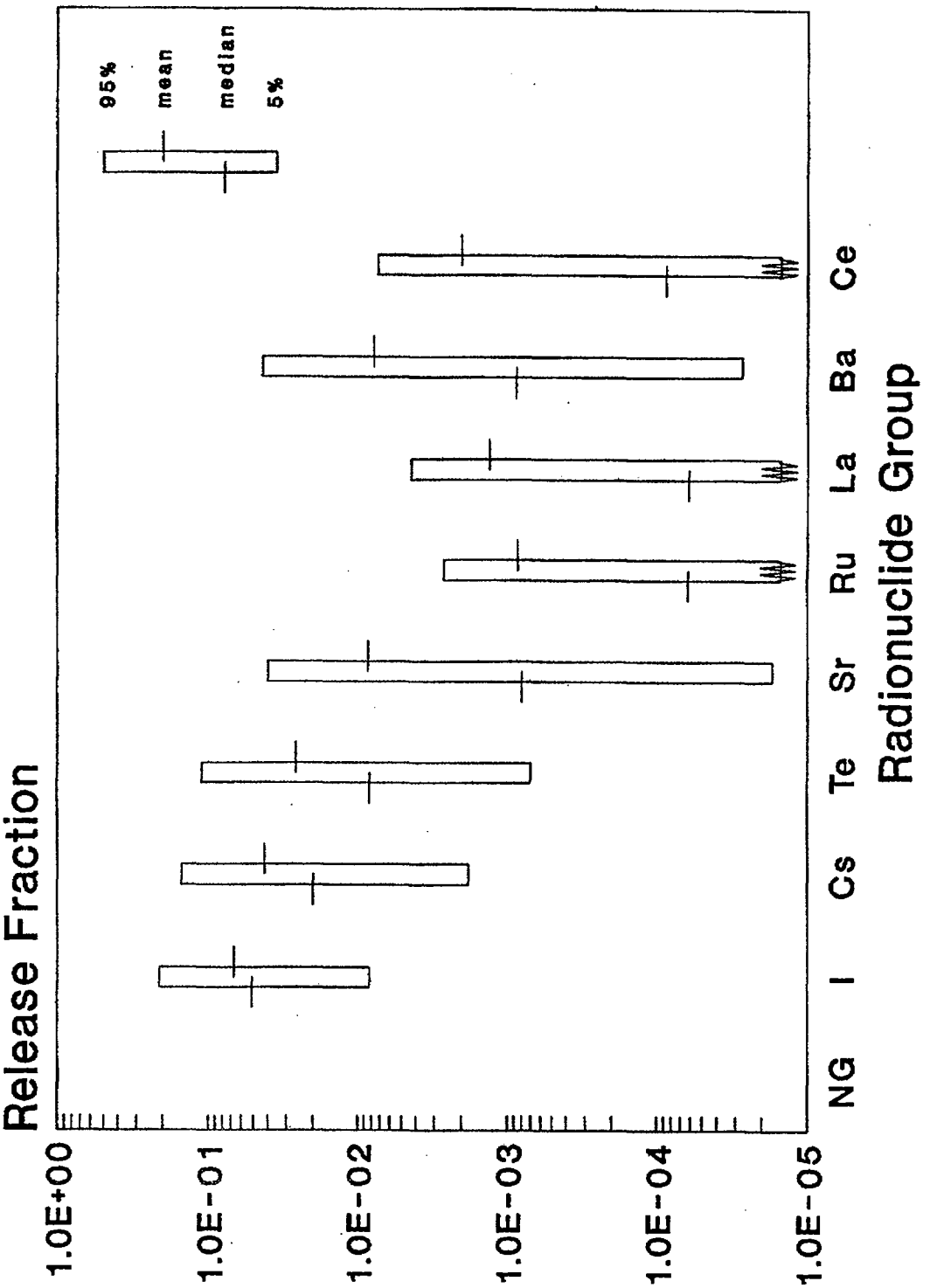


Figure 7.5 Source term distributions for early containment failure at Zion.

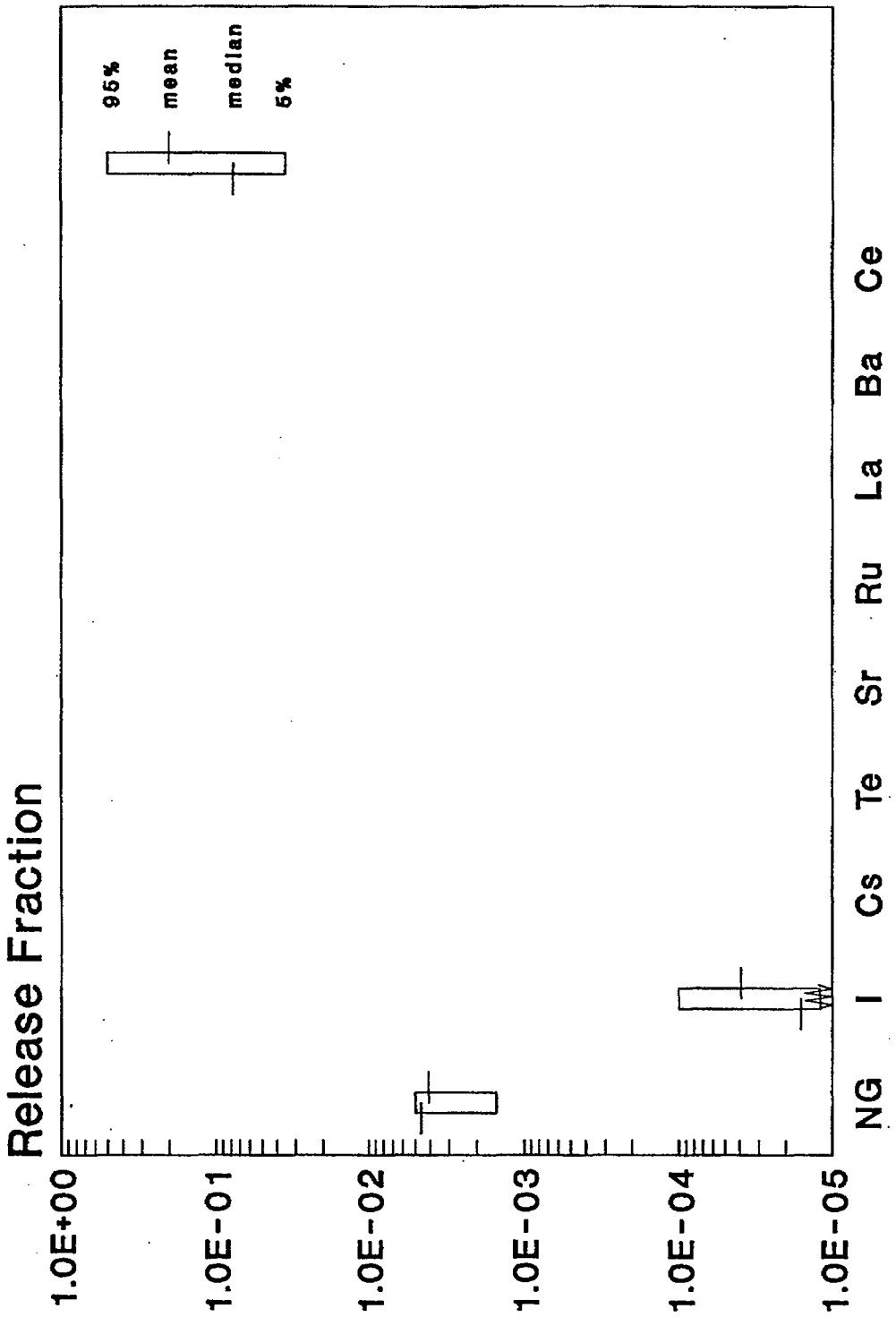


Figure 7.6 Source term distributions for no containment failure at Zion.

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:

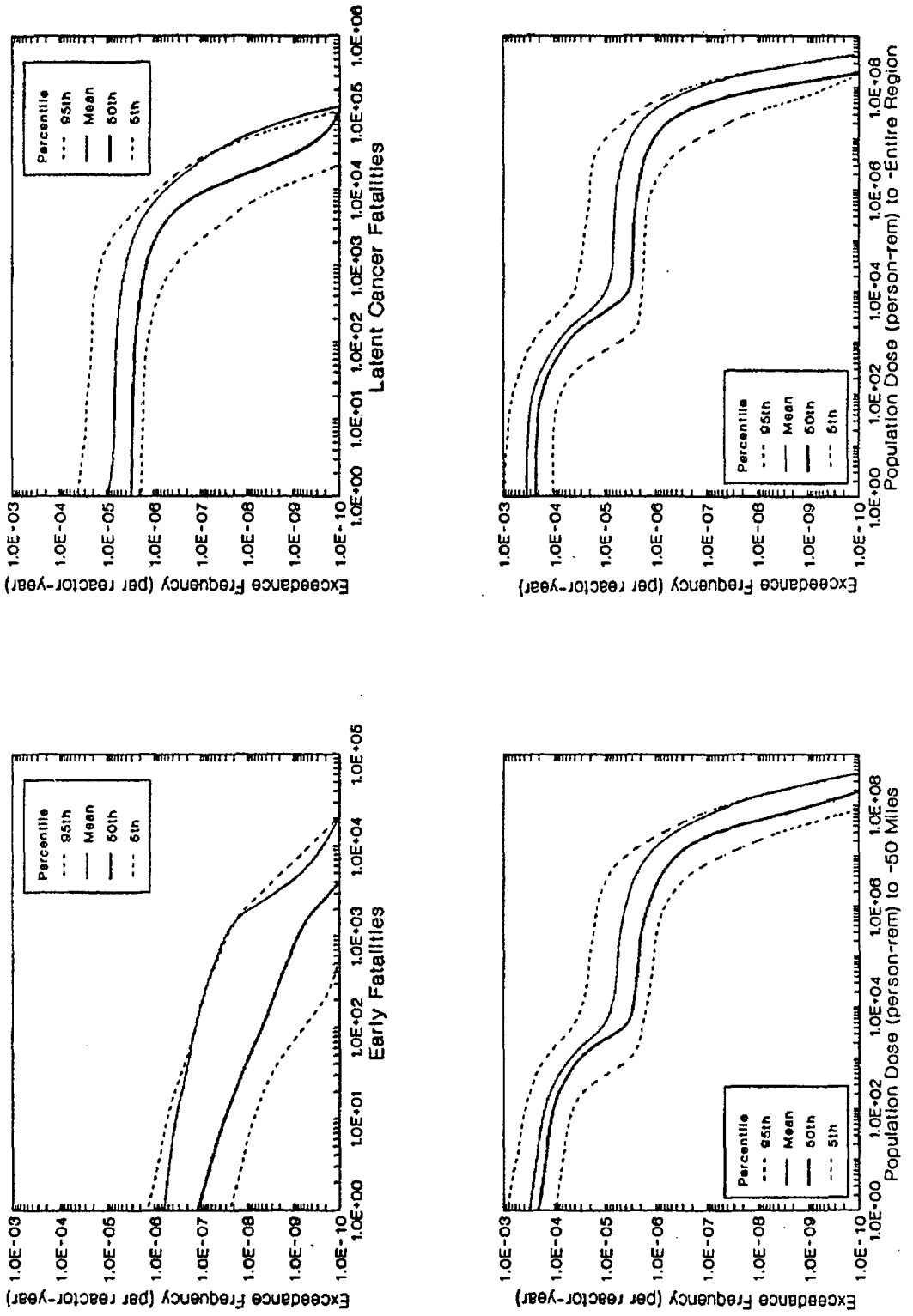
- Early fatality risk,
- Latent cancer fatality risk,
- Population dose within 50 miles of the site,
- Population dose within the entire site region,
- Individual early fatality risk in the population within 1 mile of the Zion exclusion area boundary, and
- Individual latent cancer fatality risk in the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

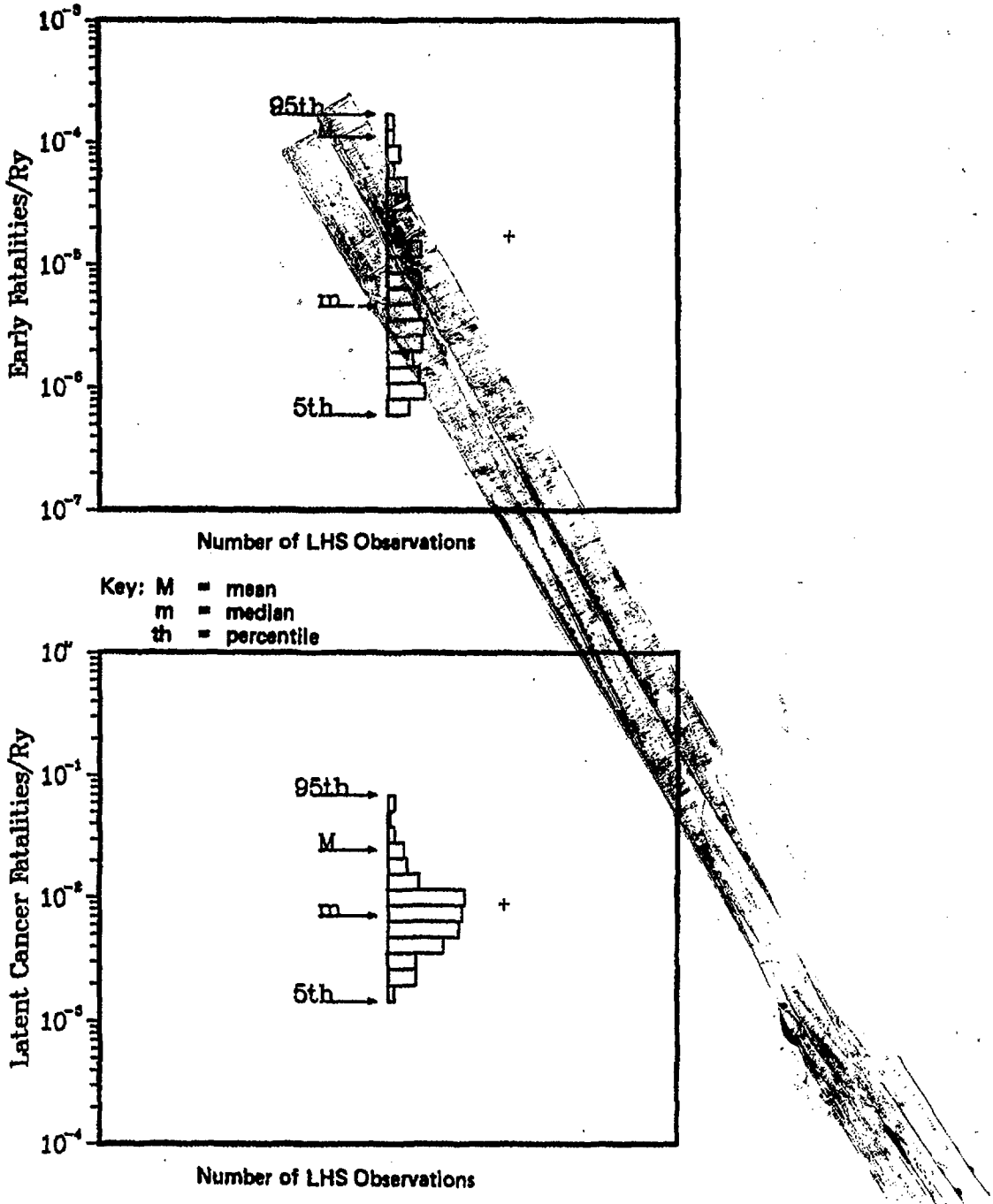
The risk results shown in Figure 7.8 have been analyzed to identify the principal contributors (accident sequences and containment failure modes) to plant risk. These results are presented in Figures 7.11 and 7.12. As may be seen, both for early and latent cancer fatality risks, the dominant plant damage 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.

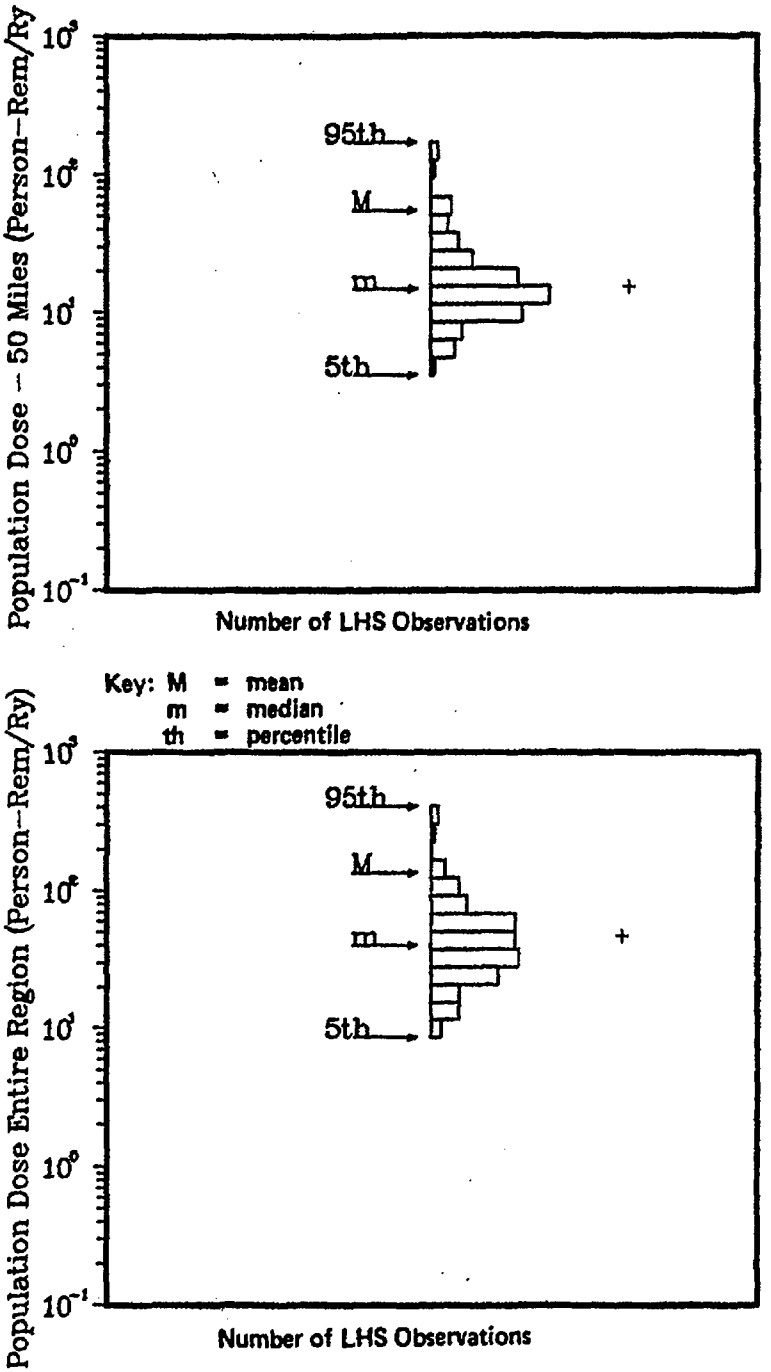
Figure 7.7 Frequency distributions of offsite consequence measures at Zion (internal initiators).



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.

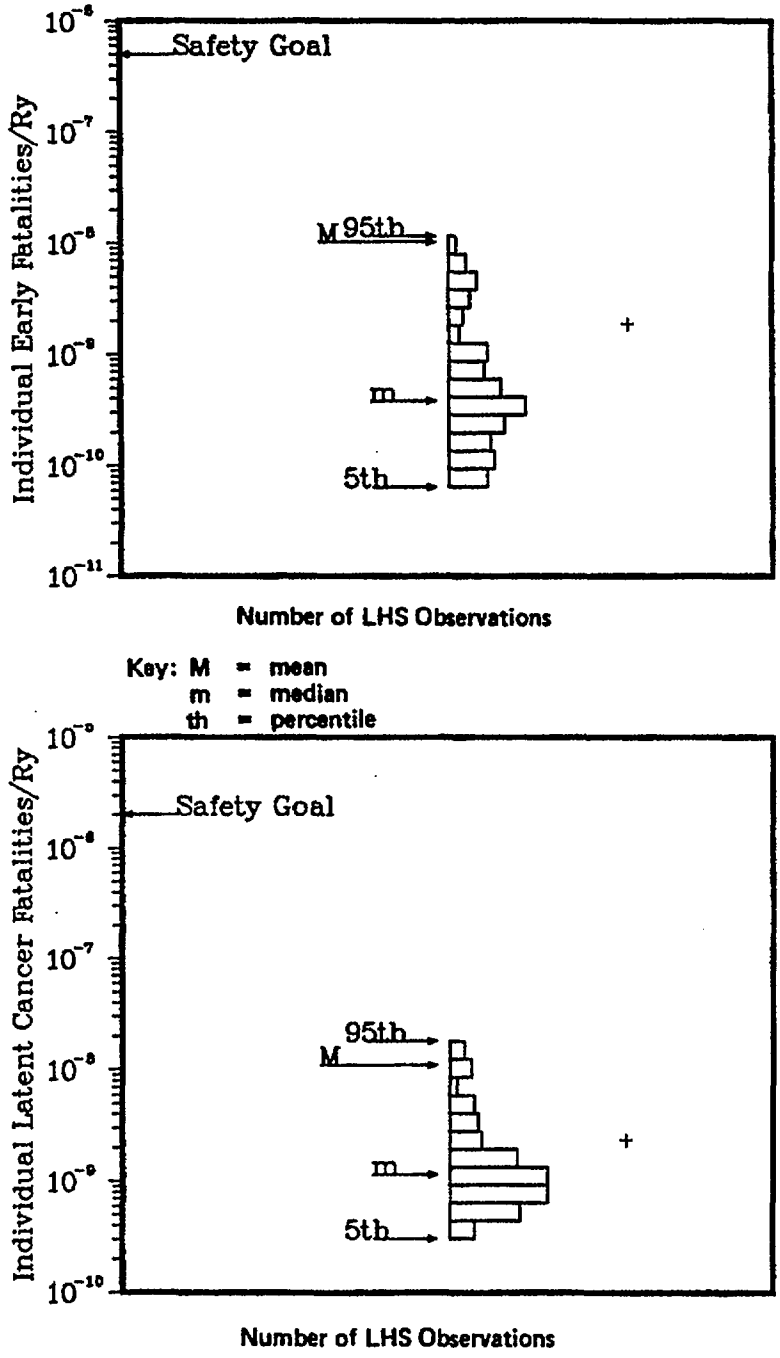
Figure 7.8 Early and latent cancer fatality risks at Zion (internal initiators).



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).

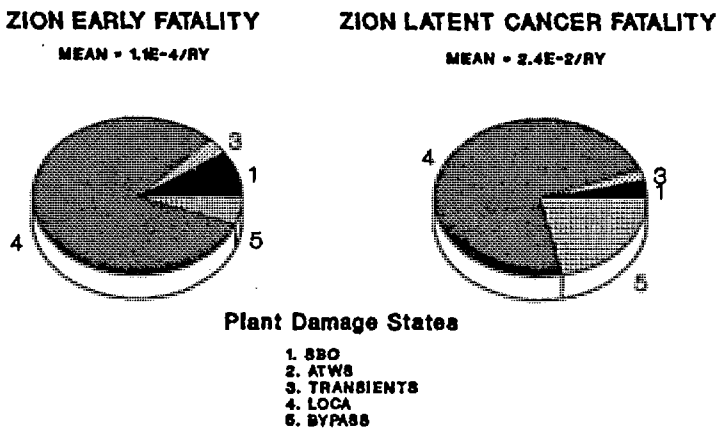


Figure 7.11 Major contributors (plant damage states) to mean early and latent cancer fatality risks at Zion (internal initiators).

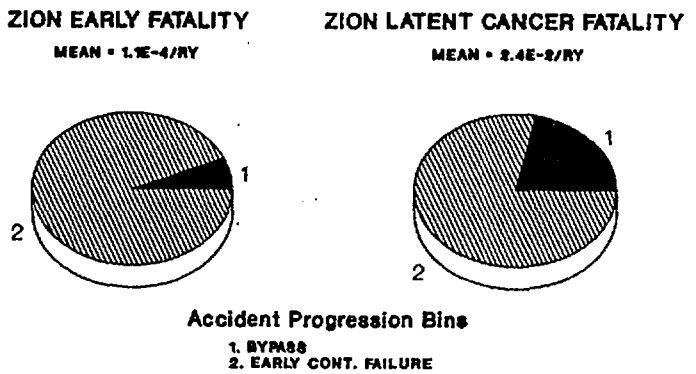


Figure 7.12 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Zion (internal initiators).

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.)

REFERENCES FOR CHAPTER 7

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

7.3 D. L. Berry et al., "Review and Evaluation of the Zion Probabilistic Safety Study: Plant Analysis," Sandia National Laboratories, NUREG/CR-3300, Vol. 1, SAND83-1118, May 1984.

7.4 Cordell Reed, Commonwealth Edison Co. (CECo), "Zion Station Units 1 and 2. Commitment to Provide a Backup Water Source to the Charging Oil Coolers," NRC Docket Nos. 50-295 and 50-304, March 13, 1989.

7.5 R. A. Chrzanowski, CECo, "March 13, 1989 Letter from Cordell Reed to T. E. Murley," NRC, NRC Docket Nos. 50-295 and 50-304, August 24, 1990.

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

7.7 Stone & Webster Engineering Corporation, "Preliminary Evacuation Time Study of the 10-Mile Emergency Planning Zone at the Zion Station," prepared for Commonwealth Edison Company, January 1980.

7.8 Federal Emergency Management Agency, "Dynamic Evacuation Analyses: Independent Assessments of Evacuation Times from the Plume Exposure Pathway Emergency Planning Zones of Twelve Nuclear Power Stations," December 1980.

7.9 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.

PART III

Perspectives and Uses

8. PERSPECTIVES ON FREQUENCY OF CORE DAMAGE

8.1 Introduction

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

8.2 Summary of Results

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

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

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

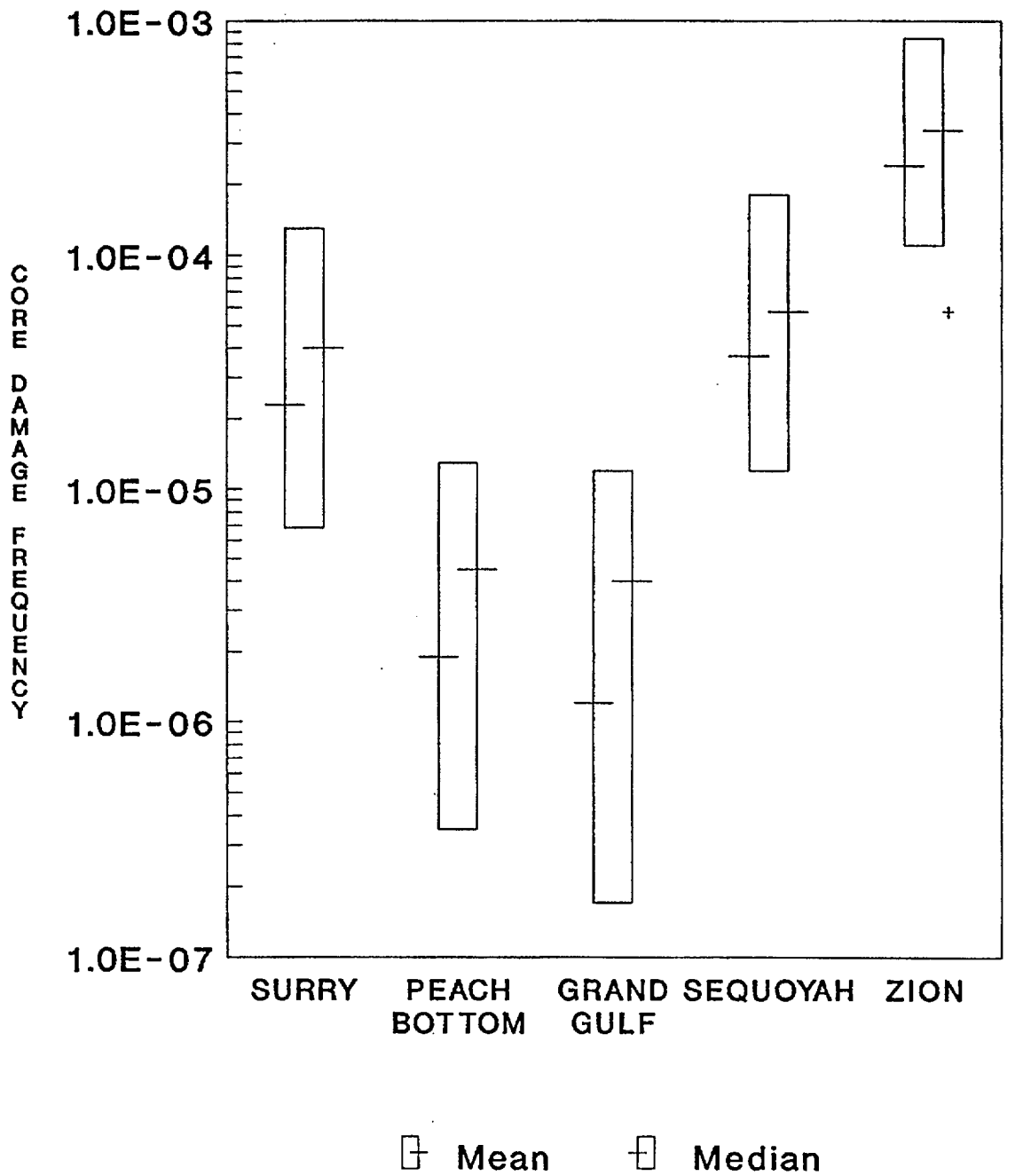
Figures 8.5 through 8.8 provide the results of the external-event analyses, and Figures 8.9 through 8.12 give the breakdown of these analyses according to the principal types of accident sequences.

8.3 Comparison with Reactor Safety Study

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

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

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



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

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

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

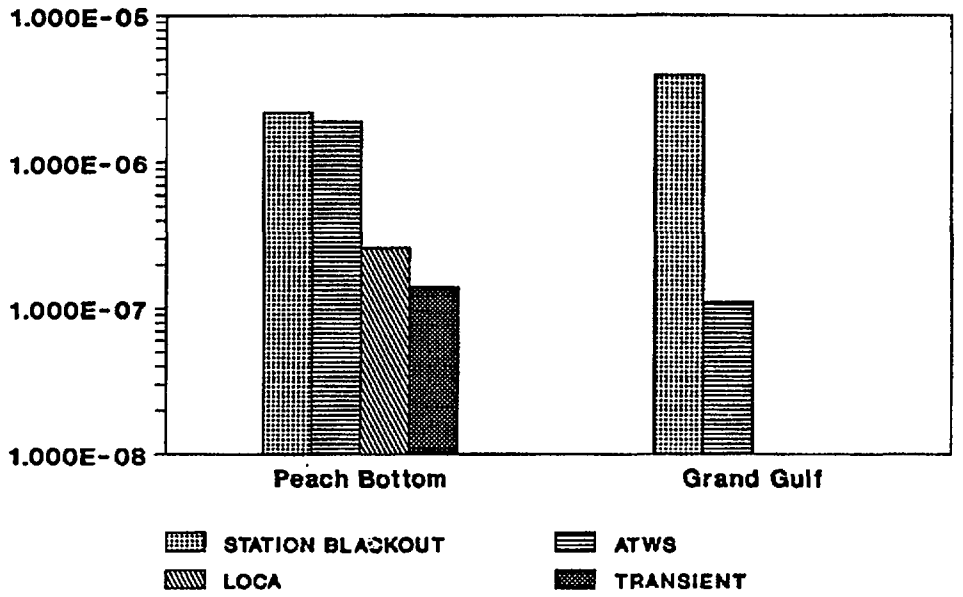
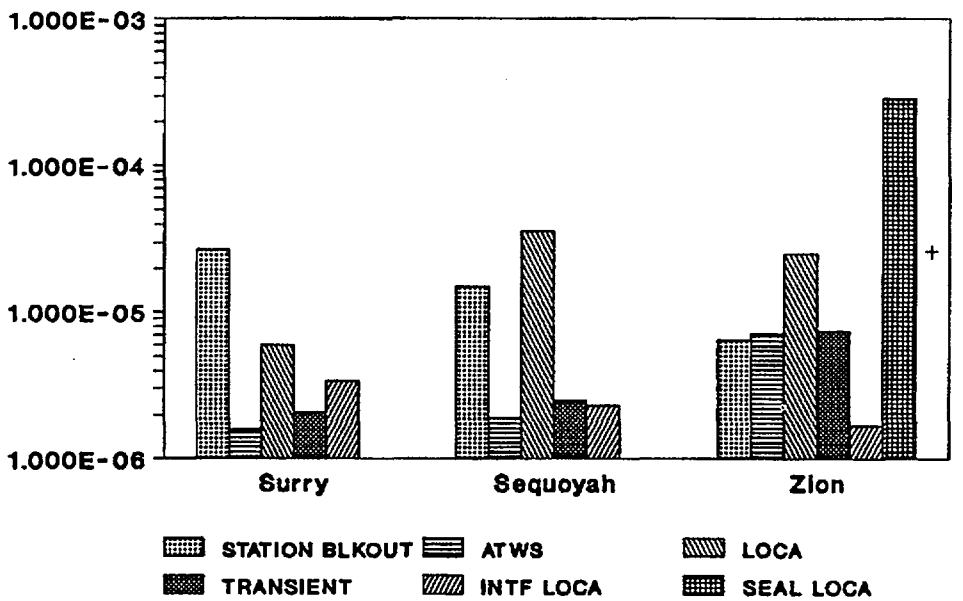


Figure 8.2 BWR principal contributors to internal core damage frequencies.



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

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

Figure 8.3 PWR principal contributors to internal core damage frequencies.

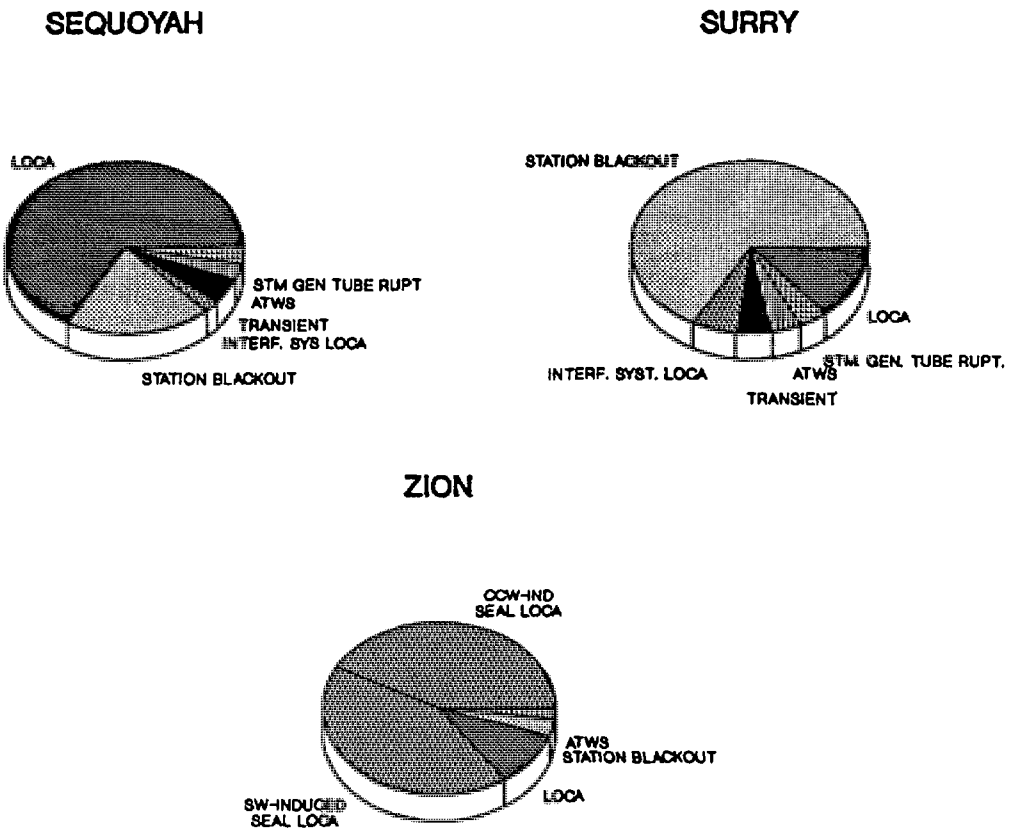


Figure 8.4 Principal contributors to internal core damage frequencies.

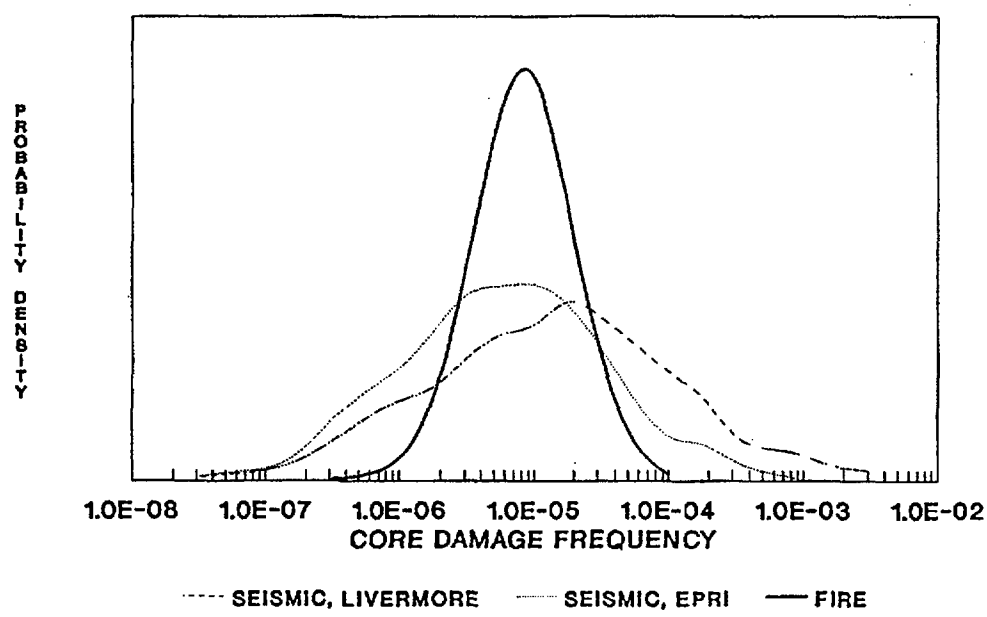
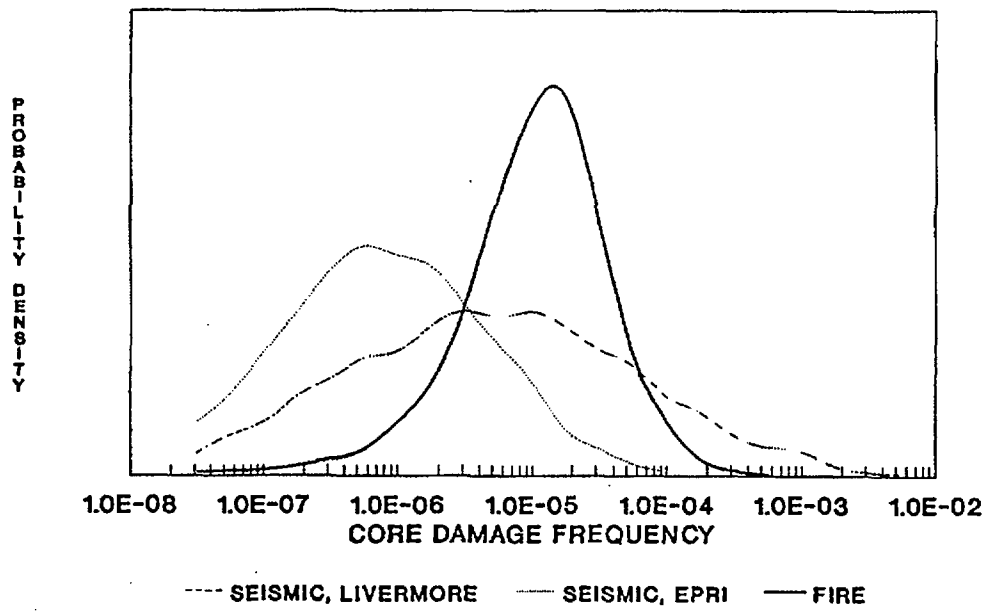


Figure 8.5 Surry external-event core damage frequency distributions.



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

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

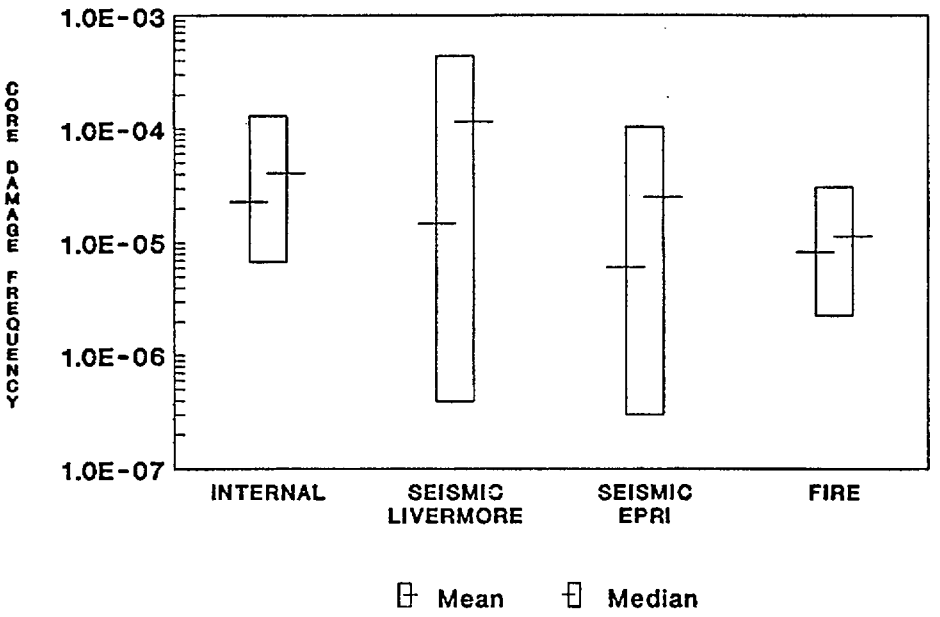
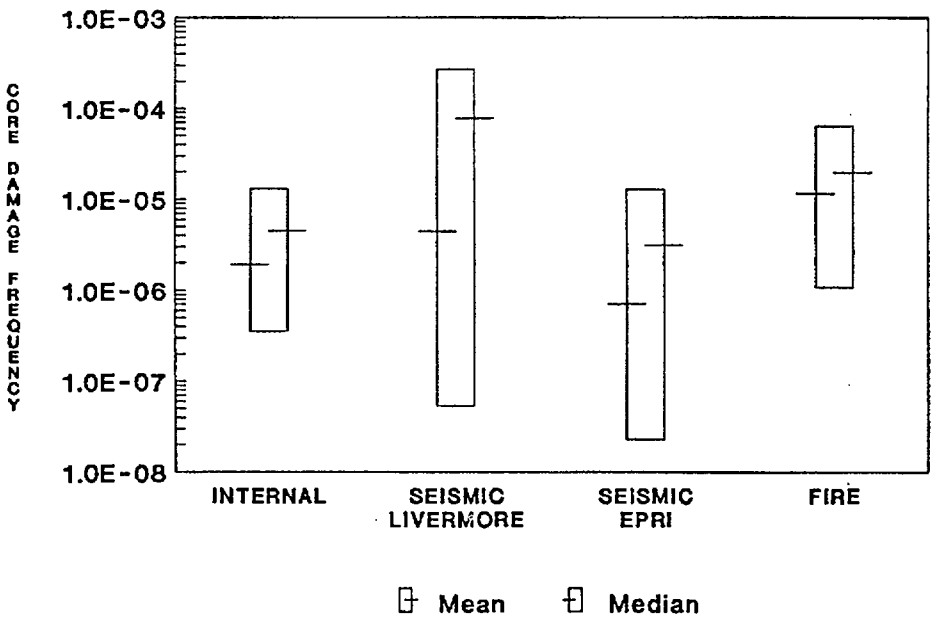


Figure 8.7 Surry internal- and external-event core damage frequency ranges.



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

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

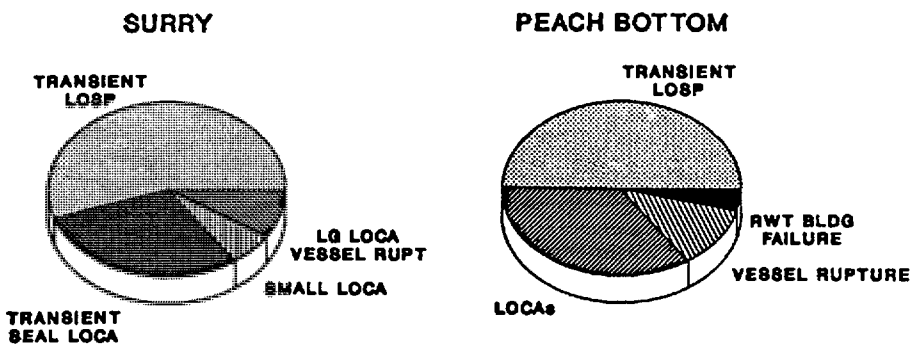


Figure 8.9 Principal contributors to seismic core damage frequencies.

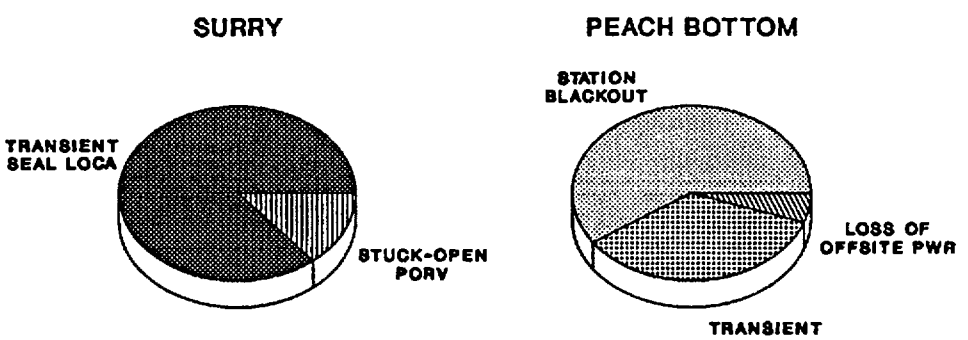


Figure 8.10 Principal contributors to fire core damage frequencies.

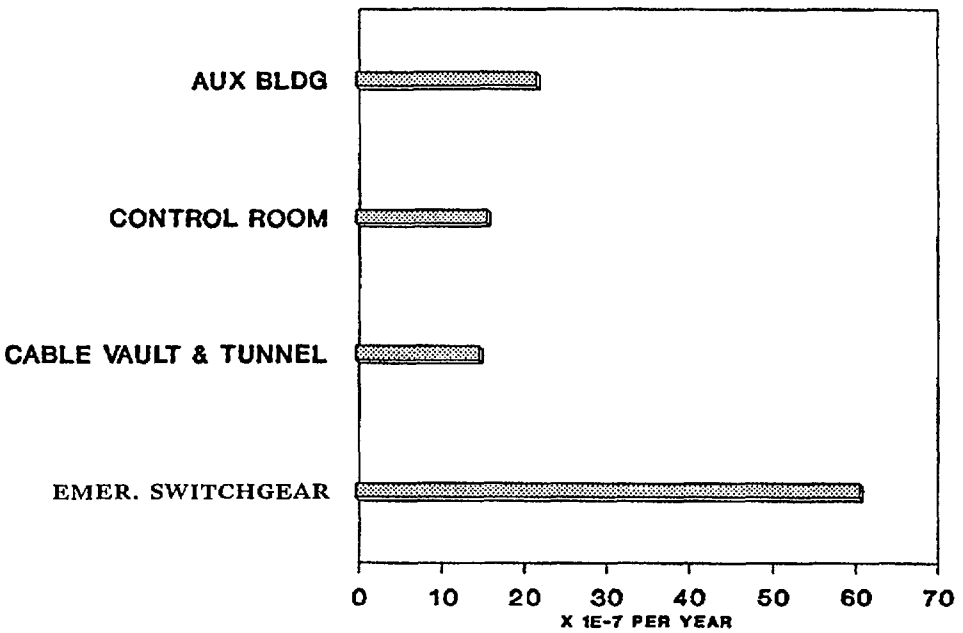
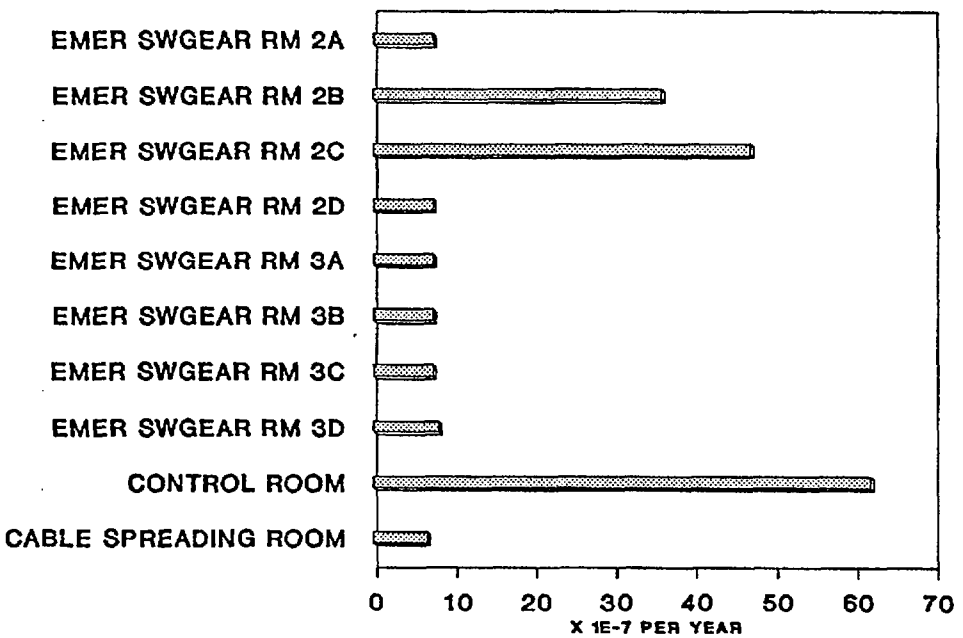


Figure 8.11 Surry mean fire core damage frequency by fire area.



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

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

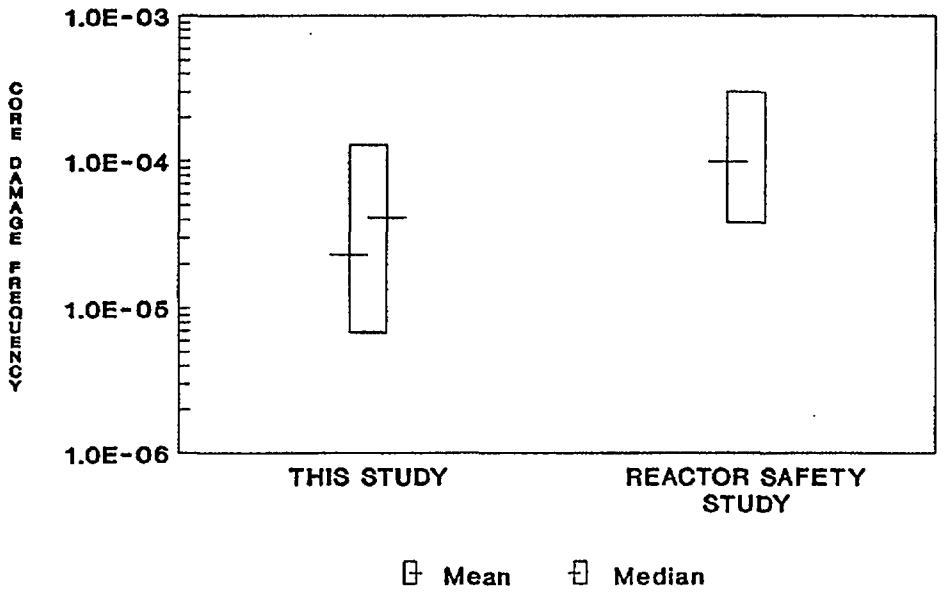
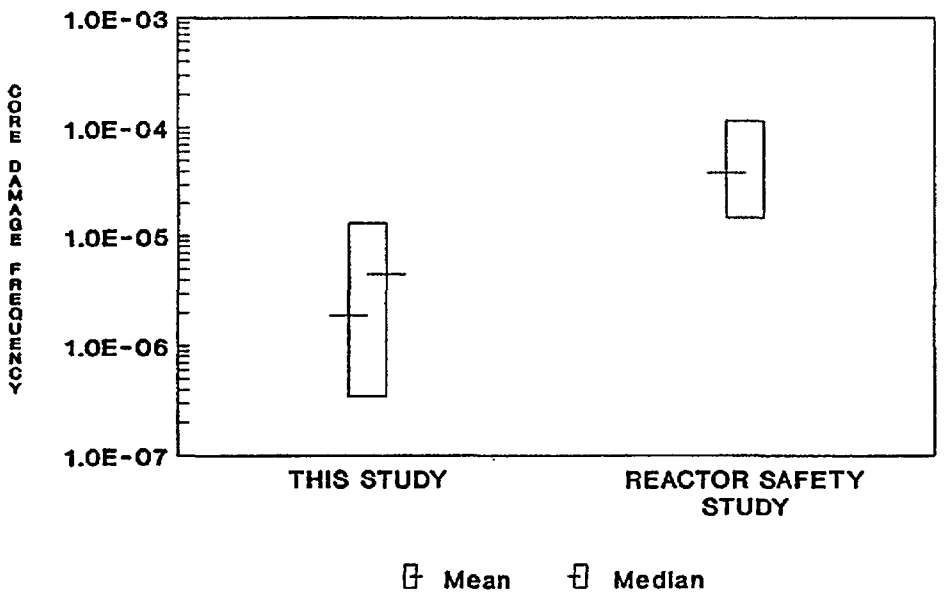


Figure 8.13 Comparison of Surry internal core damage frequency with Reactor Safety Study.



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

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

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

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

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

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

8.4 Perspectives

8.4.1 Internal-Event Core Damage Probability Distributions

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

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

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

frequency between Surry and Sequoyah are not very significant.

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

8.4.2 Principal Contributors to Uncertainty in Core Damage Frequency

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

8.4.3 Dominant Accident Sequence Types

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

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

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

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

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

Boiling Water Reactor versus Pressurized Water Reactor

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

The LOCA sequences, often dominant in the PWR core damage frequencies, are minor contributors in the case of the BWRs. This is not surprising in view of the fact that most BWRs have many more systems than PWRs for injecting water

directly into the reactor coolant system to provide makeup. For BWRs, this includes two low-pressure emergency core cooling (ECC) systems (low-pressure coolant injection and low-pressure core spray), each of which is multitrain; two high-pressure injection systems (reactor core isolation cooling and either high-pressure coolant injection or high-pressure core spray); and usually several other alternative injection systems, such as the control rod drive hydraulic system, condensate, service water, firewater, etc. In contrast, PWRs generally have one high-pressure and one low-pressure ECC system (both multitrain), plus a set of accumulators. The PWR ECCS does have considerable redundancy, but not as much as that of most BWRs.

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

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

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

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

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

Boiling Water Reactor Observations

As shown in Figure 8.1, the internal-event core damage frequencies for Peach Bottom and Grand Gulf are extremely low. Therefore, even though dominant plant damage states and contributing

failure events can be identified, these items should not be considered as safety problems for the two plants. In fact, these dominating factors should not be overemphasized because, for core damage frequencies below 1E-5, it is possible that other events outside the scope of these internal-event analyses are the ones that actually dominate. In the cases of these two plants, the real perspectives come not from understanding why particular sequences dominate, but rather why all types of sequences considered in the study have low frequencies for these plants.

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

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

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

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

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

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

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

positive suction head in the suppression pool, and also the reactor building is subjected to a harsh steam environment that can lead to failure of equipment located there.

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

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

Pressurized Water Reactor Observations

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

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

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

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

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

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

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

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

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

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

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

8.4.4 External Events

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

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

Seismic Analysis Observations

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

The sciences of geology and seismology have not yet produced a model or group of models upon which all experts agree. This study did not itself

produce seismic hazard curves, but instead made use of seismic hazard curves for Peach Bottom and Surry that were part of an NRC-funded Lawrence Livermore National Laboratory project that resulted in seismic hazard curves for all nuclear power plant sites east of the Rocky Mountains (Ref. 8.9).

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

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

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

1. Surry Seismic Analysis

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

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

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

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

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

2. Peach Bottom Seismic Analysis

As can be seen in Figure 8.9, the dominant contributor in the seismic core damage frequency analysis is a transient sequence brought about by loss of offsite power. The loss of offsite power is due to seismically induced failures of onsite ac power. Peach Bottom has four emergency diesel

generators, all shared between the two units, and four station batteries per unit. Thus, there is a high degree of redundancy. However, all diesels require cooling provided by the emergency service water system, and failure to provide this cooling will result in failure of all four diesels.

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

Fire Analysis Observations

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

1. Surry Fire Analysis

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

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

2. Peach Bottom Fire Analysis

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

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

The cable spreading room below the control room is significant but not dominant in the fire analysis. The scenario of interest is a fire-induced transient coupled with fire-related failures of the control power for the high-pressure coolant injection

system, the reactor core isolation cooling system, the automatic depressurization system, and the control rod drive hydraulic system. The analysis gave credit to the automatic CO_2 fire-suppression system in this area.

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

Perspectives: General Observations on Fire Analysis

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

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

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

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9. PERSPECTIVES ON ACCIDENT PROGRESSION AND CONTAINMENT PERFORMANCE

9.1 Introduction

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

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

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

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

9.2 Summary of Results

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

Not all accident sequences that involve core damage would necessarily progress to vessel failure, as illustrated by the TMI-2 accident. The operator may recover a critical system (such as by the return of offsite power) or the state of the plant may change (for example, the system pressure may fall to a point where low-pressure emergency coolant

systems can be activated) allowing the core to be recovered and the accident to be terminated. The likelihood of containment failure in terminated accidents is typically less than in accidents involving vessel failure, and the radiological consequences are usually very small.

9.2.1 Internal Events

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

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

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

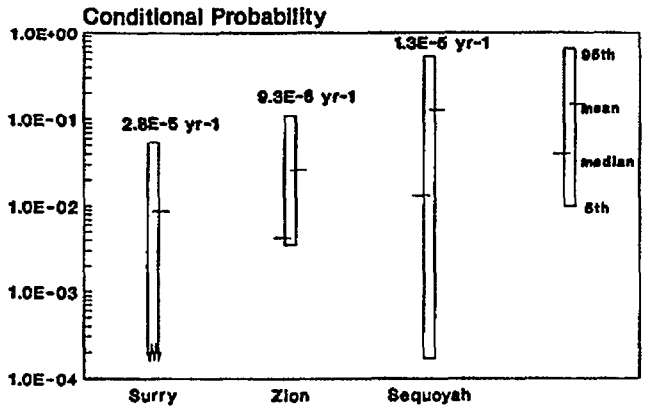
For accident sequences at Surry and Zion in which core uncovery is initiated with the reactor coolant system at high pressure, the probability of overheating and rupturing steam generator tubes after the onset of core damage, with subsequent bypass of the containment, is of the same magnitude as the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

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

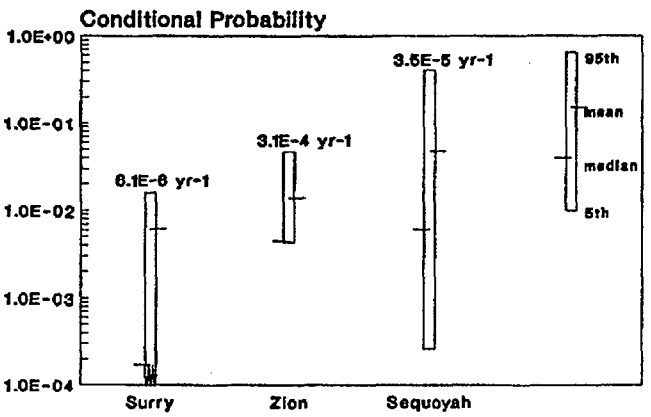
The predicted probability of early failure of the Peach Bottom and Grand Gulf pressure-suppression containments is substantially higher than for the PWR containment designs. For Grand Gulf, the mean probability of early failure is approximately 50 percent while at Peach Bottom the mean probability of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

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

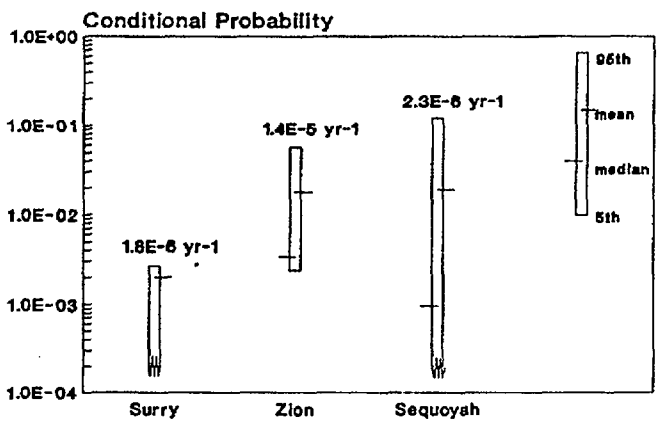
At Grand Gulf, early containment failure in station blackout is dominated by hydrogen deflagrations. Hydrogen detonations are also small contributors to early failure. For short-term station blackouts (the dominant plant damage state groups), the conditional probability of early



a. Station blackout

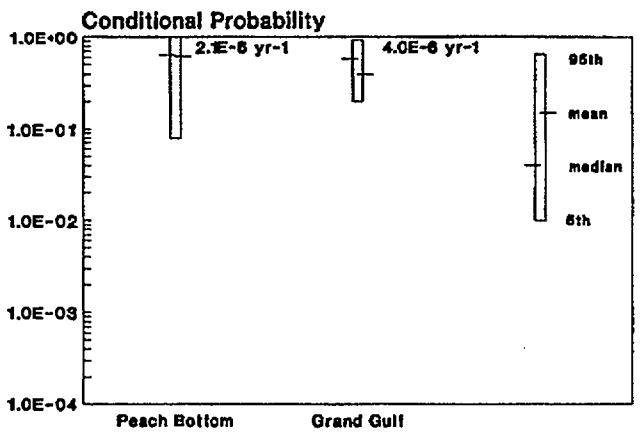


b. Loss-of-coolant accidents

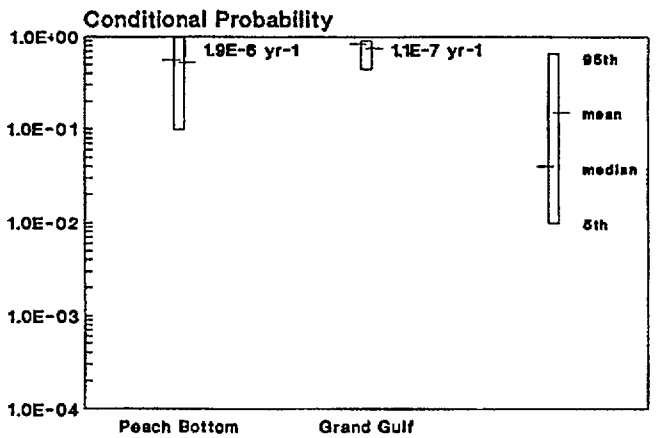


c. Transients

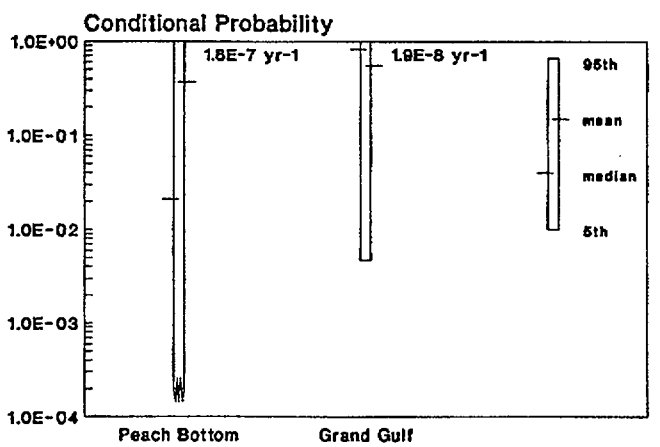
Figure 9.1 Conditional probability of early containment failure for key plant damage states (PWRs).



a. Station blackout



b. Anticipated transients without scram



c. Transients

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

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

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

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

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

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

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

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

9.2.2 External Events

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

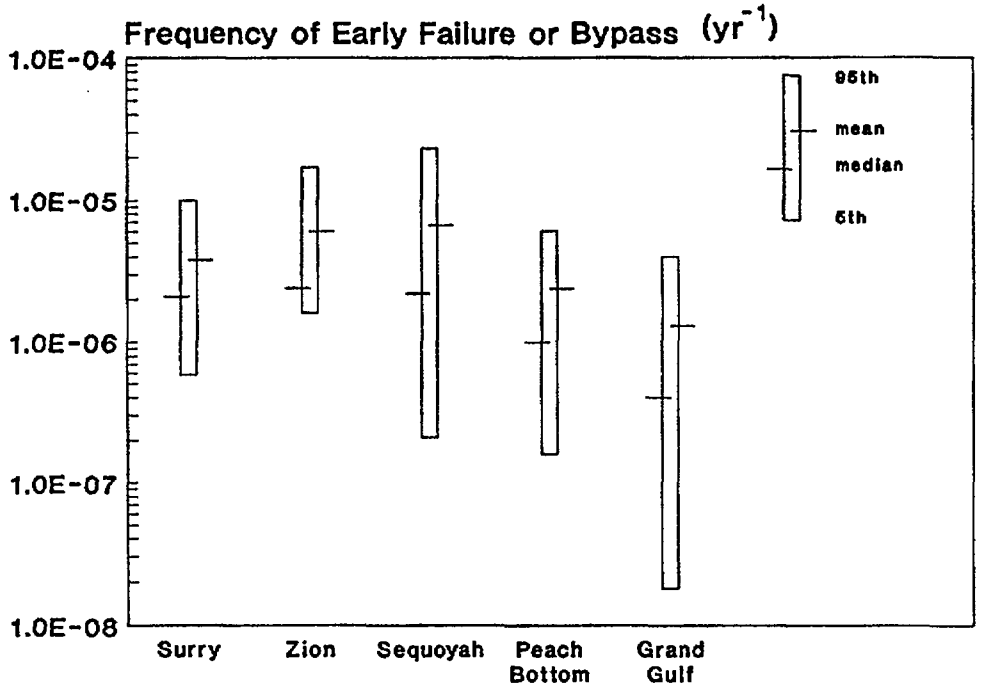


Figure 9.3 Frequency of early containment failure or bypass (all plants).

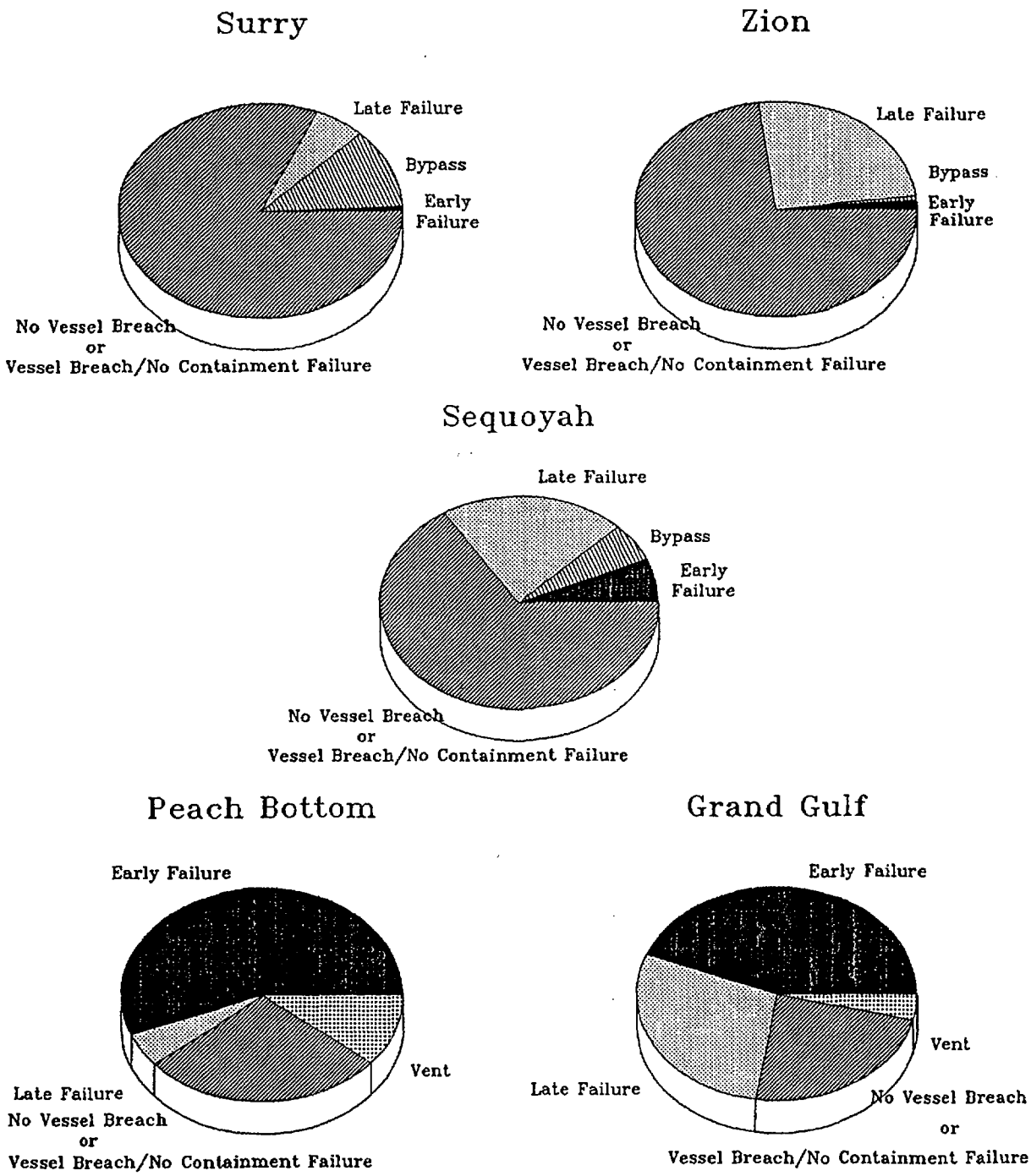


Figure 9.4 Relative probability of containment failure modes (internal events).

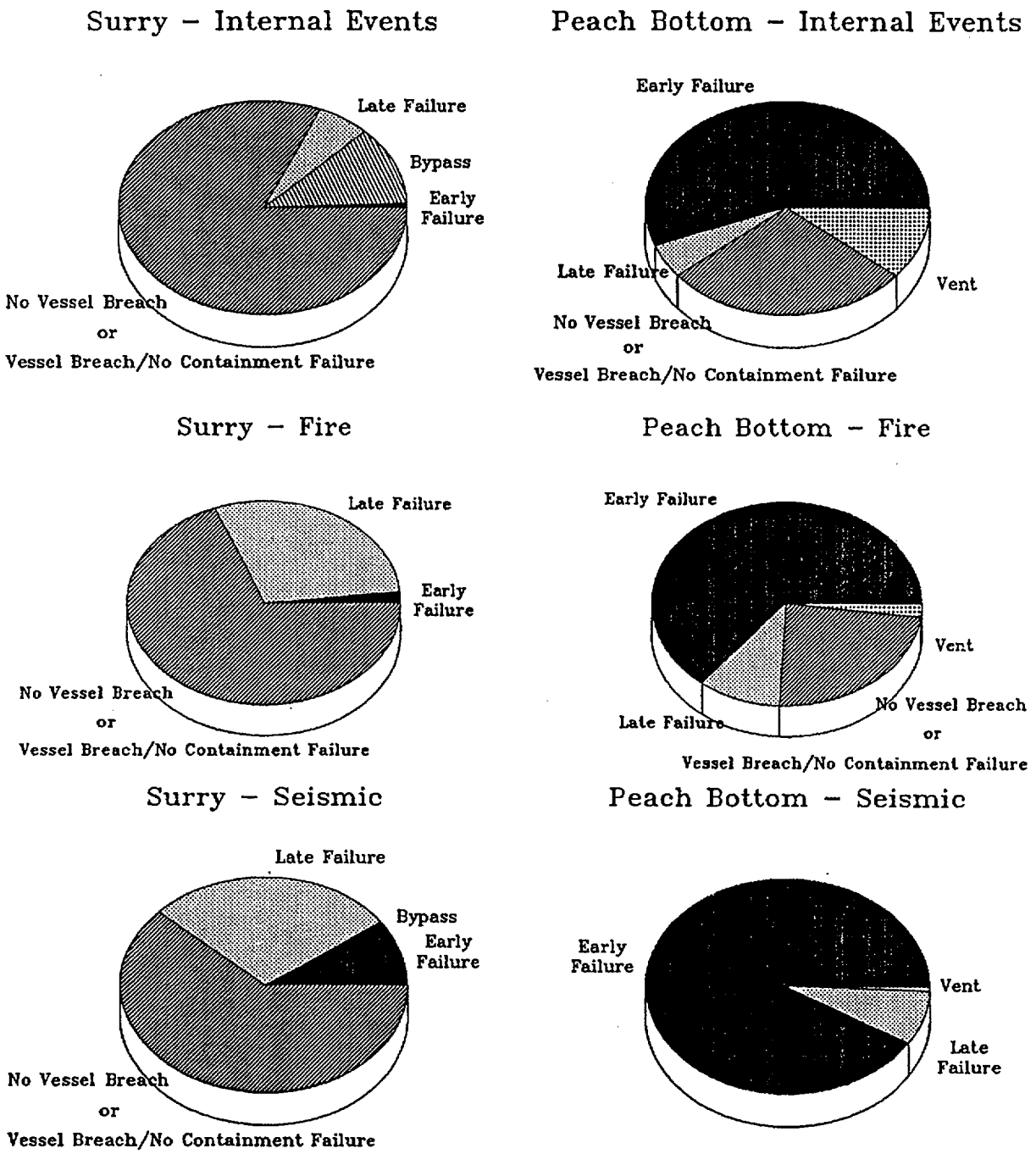


Figure 9.5 Relative probability of containment failure modes (internal and external events, Surry and Peach Bottom).

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

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

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

9.2.3 Additional Summary Results

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

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

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

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

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

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

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

Sequoyah Plant (Ice Condenser Design)

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

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

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

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

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

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

Peach Bottom Plant (Mark I Design)

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

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

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

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

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

Grand Gulf Plant (Mark III Design)

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

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

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

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

9.3 Comparison with Reactor Safety Study

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

Containment Failure Modes

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

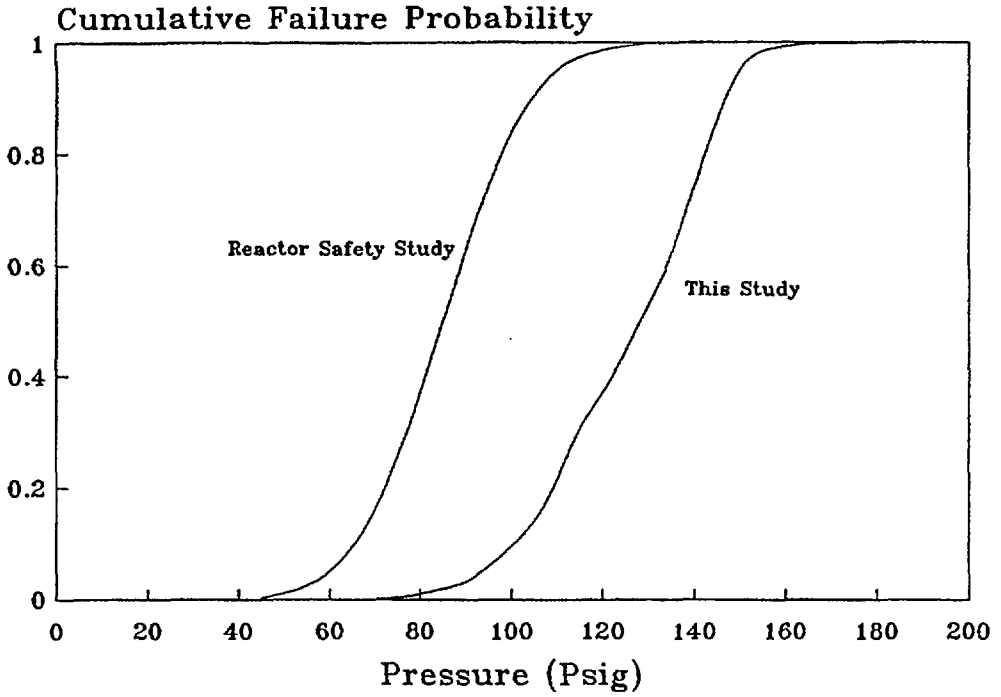


Figure 9.6 Comparison of containment failure pressure with Reactor Safety Study (Surry).

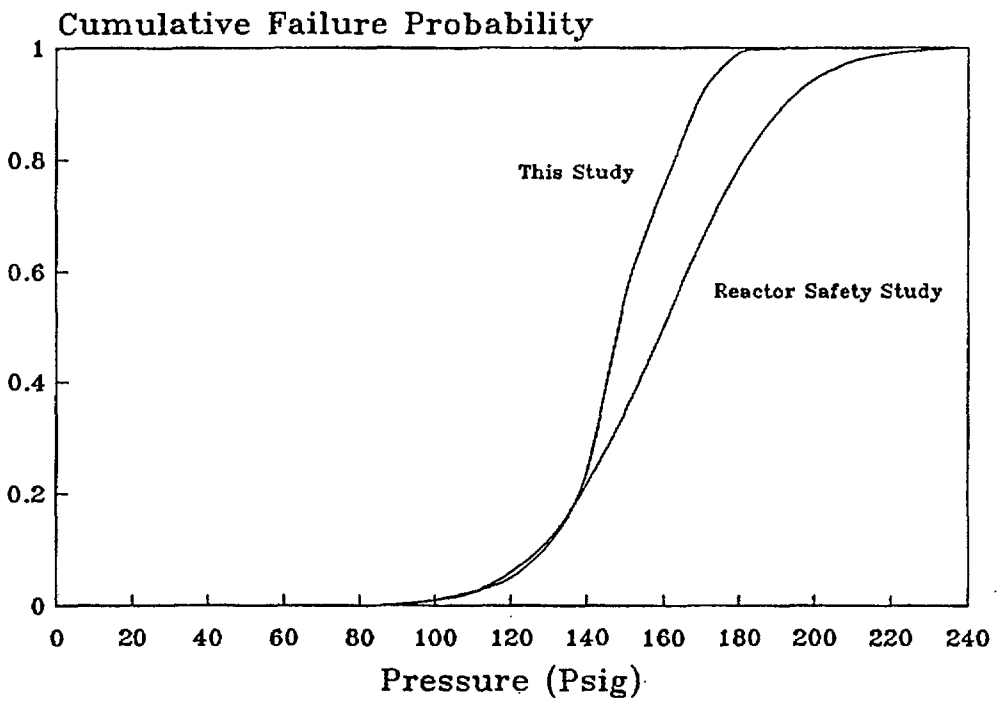


Figure 9.7 Comparison of containment failure pressure with Reactor Safety Study (Peach Bottom).

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

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

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

Comparison of Surry Results

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

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

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

Comparison of Peach Bottom Results

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

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

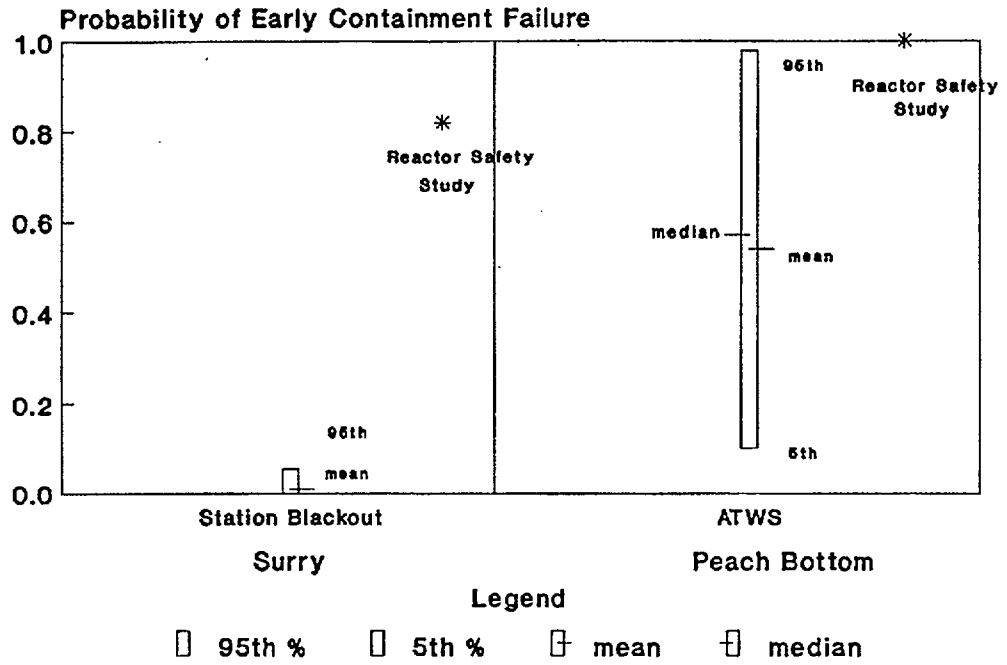


Figure 9.8 Comparison of containment performance results with Reactor Safety Study (Surry and Peach Bottom).

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

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

9.4 Perspectives

9.4.1 State of Analysis Methods

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

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

Computer analyses cannot, in general, be used directly and alone to calculate branching probabilities in the accident progression event tree. Since the greatest source of uncertainty is typically associated with the modeling of severe accident

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

9.4.2 Important Mechanisms That Defeat Containment Function During Severe Accidents

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

Containment Bypass and Isolation Failure

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

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

Gas Combustion

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

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

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

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

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

short-term station blackout sequences resulting in condensation of the steam from the drywell or the vessel and leading to hydrogen-rich mixtures in the containment that are readily ignited.

Loads at Vessel Failure

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

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

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

Direct Attack by Molten Debris

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

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

Steam Explosions

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

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

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

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

Gradual Overpressurization

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

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

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

Overheating

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

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

Loss of Vessel Support

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

Basemat Meltthrough

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

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

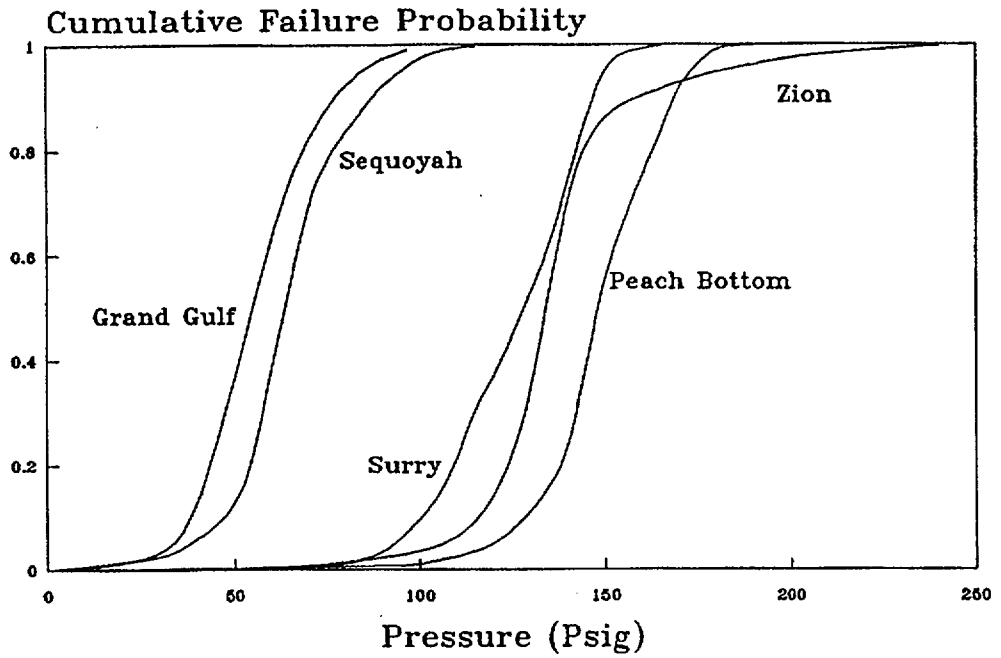


Figure 9.9 Cumulative containment failure probability distribution for static pressurization (all plants).

9.4.3 Major Sources of Uncertainty

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

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

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

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

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

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

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

The regression analyses performed for Sequoyah showed the containment failure pressure and

loads at vessel breach to be accident progression variables somewhat important to the uncertainty in both total early fatality risk and total latent cancer fatality risk (Ref. 9.4).

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

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

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

Direct Attack of Drywell Shell in Peach Bottom

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

High-Pressure Melt Ejection and Vessel Depressurization

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

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

hydrogen produced in the oxidation of dispersed core debris by steam.

Containment Failure by Steam Explosions

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

Core Melt Progression

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

Containment Bypass

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

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10. PERSPECTIVES ON SEVERE ACCIDENT SOURCE TERMS

10.1 Introduction

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

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

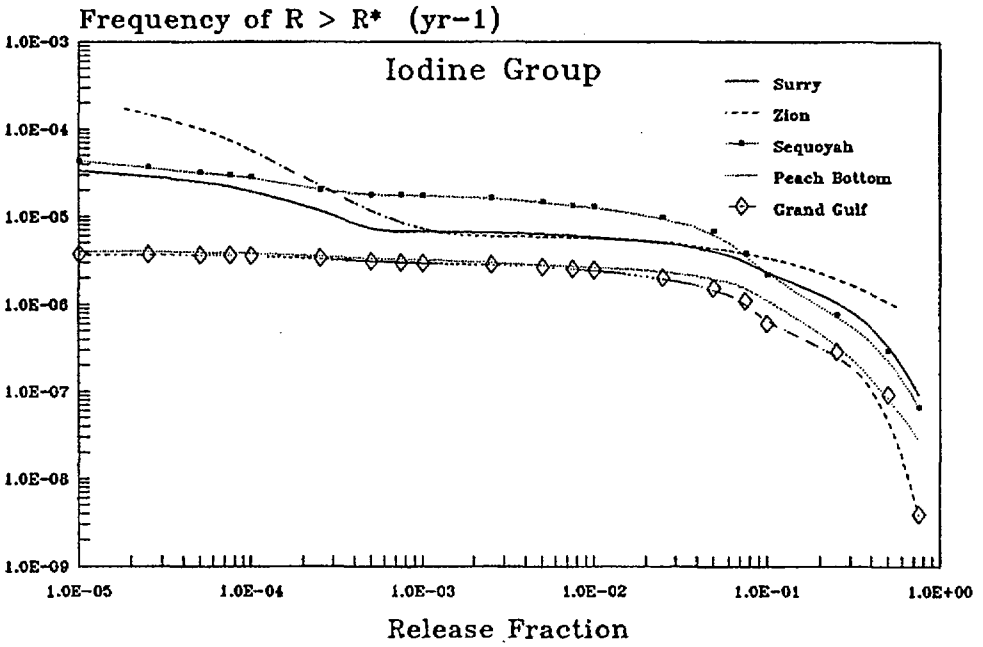
It is widely believed that the approximate treatment of source term phenomena in the Reactor Safety Study (RSS) (Ref. 10.7) analyses led to a substantial overestimation of severe accident consequences and risk. The current risk analyses provide a basis for understanding the differences that exist in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

10.2 Summary of Results

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

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

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



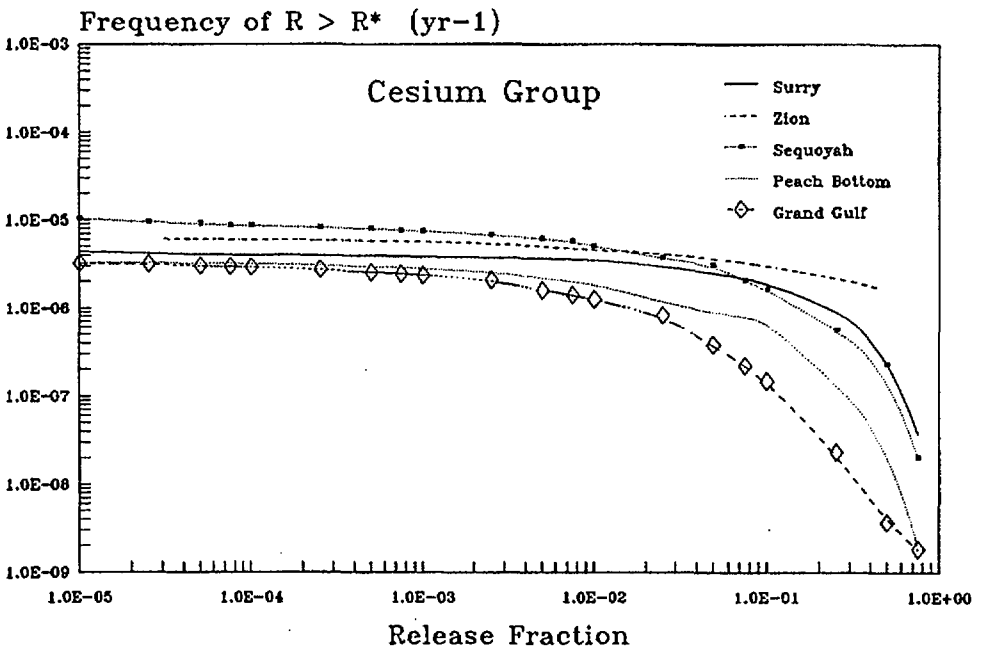
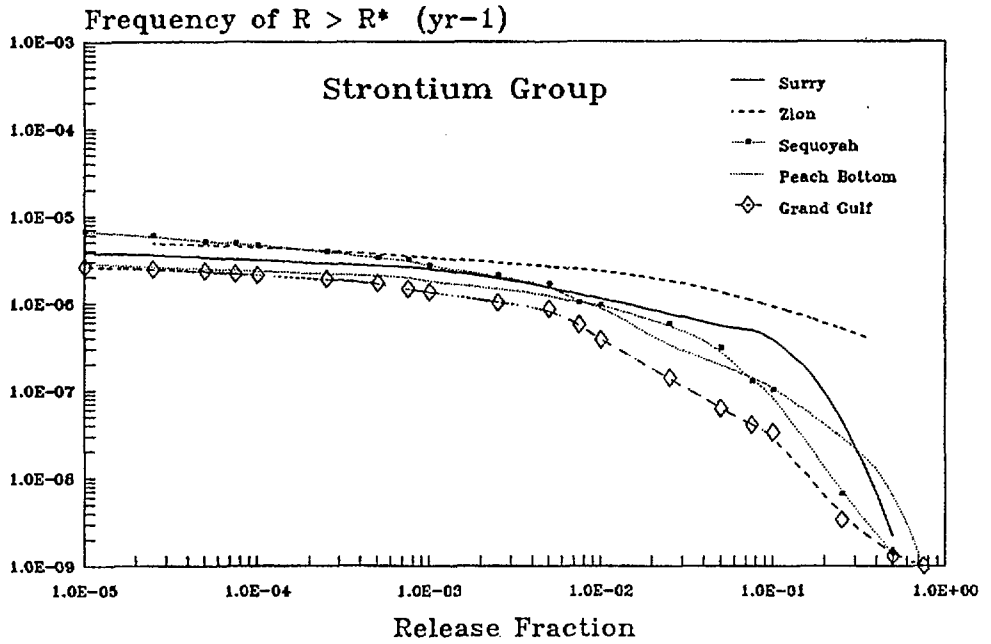


Figure 10.1 Frequency of release for key radionuclide groups.



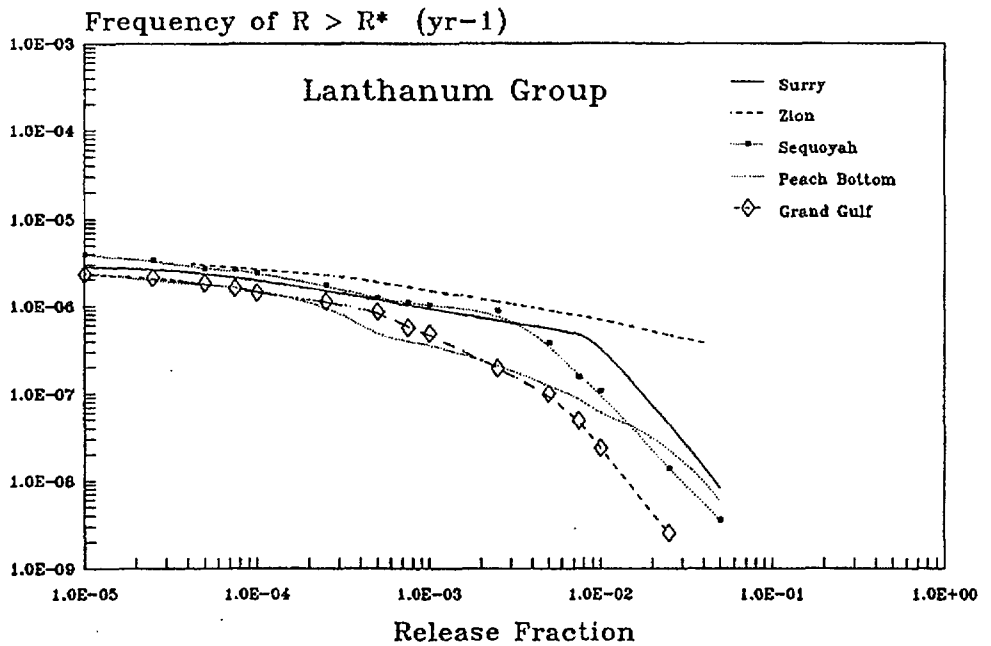


Figure 10.1 (Continued)

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

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

- The uncertainty in radionuclide source terms is large and represents a significant contribution to the uncertainty in the absolute value of risk. The relative significance of source term uncertainties depends on the plant damage state.

- Source terms for bypass sequences, such as accidents initiated by steam generator tube rupture (SGTR), can be quite large, potentially comparable to the largest Reactor Safety Study source terms.

- Early containment failure by itself is not a reliable indicator of the severity of severe accident source terms. Substantial retention of radionuclides is predicted to occur in many of the early containment failure scenarios in the BWR pressure-suppression designs, particularly for the in-vessel period of release during which radionuclides are transported to the suppression pool. Containment spray system and ice condenser decontamination can also substantially mitigate accident source terms.

- Flooding of reactor cavities or pedestals can eliminate the core-concrete release of radionuclides, if a coolable debris bed is formed, or can significantly attenuate the release from the molten core-concrete interaction by scrubbing in the overlaying pool of water.

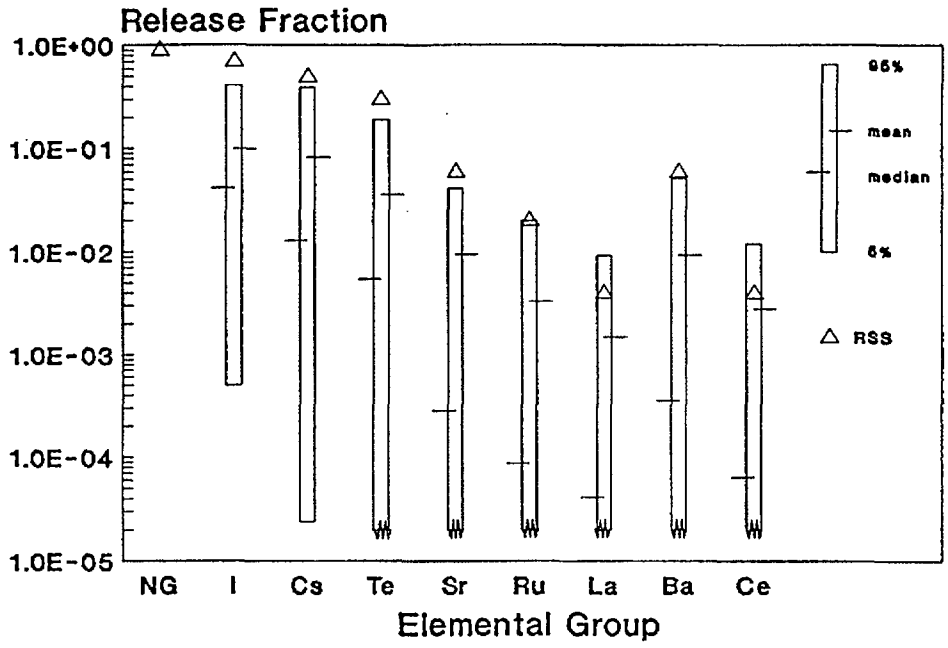
10.3 Comparison with Reactor Safety Study

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

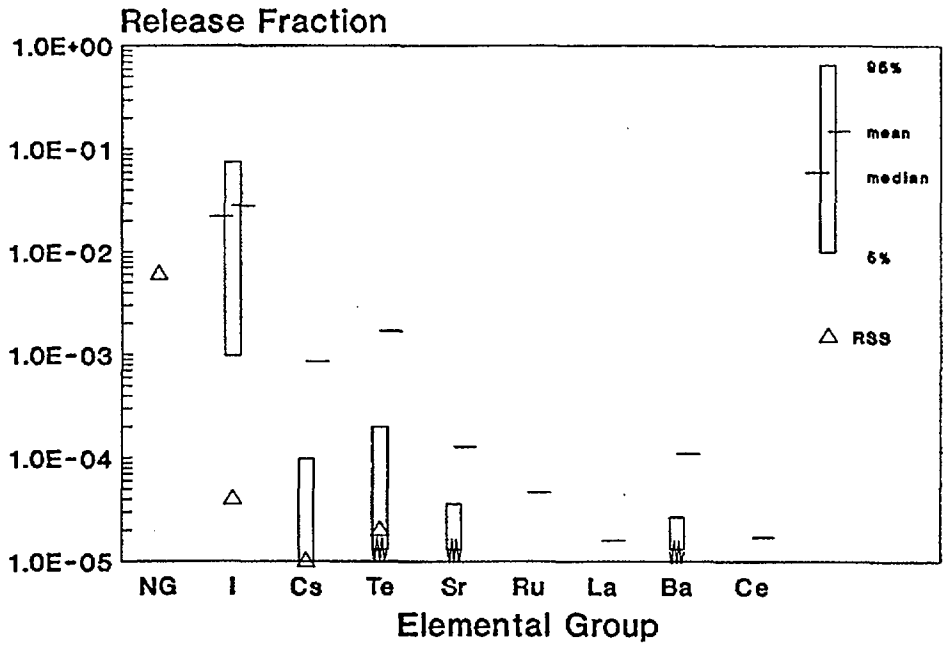
In the current study, source terms were developed for a much larger number of accident progression bins. A distribution of release fractions was also obtained for each of the elemental groups corresponding to the individual sample members of the uncertainty analysis.

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

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



a. Comparison with Bin PWR2



b. Comparison with Bin PWR7

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

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

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

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

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

10.4 Perspectives

10.4.1 State of Methods

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

Improvement in Understanding

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

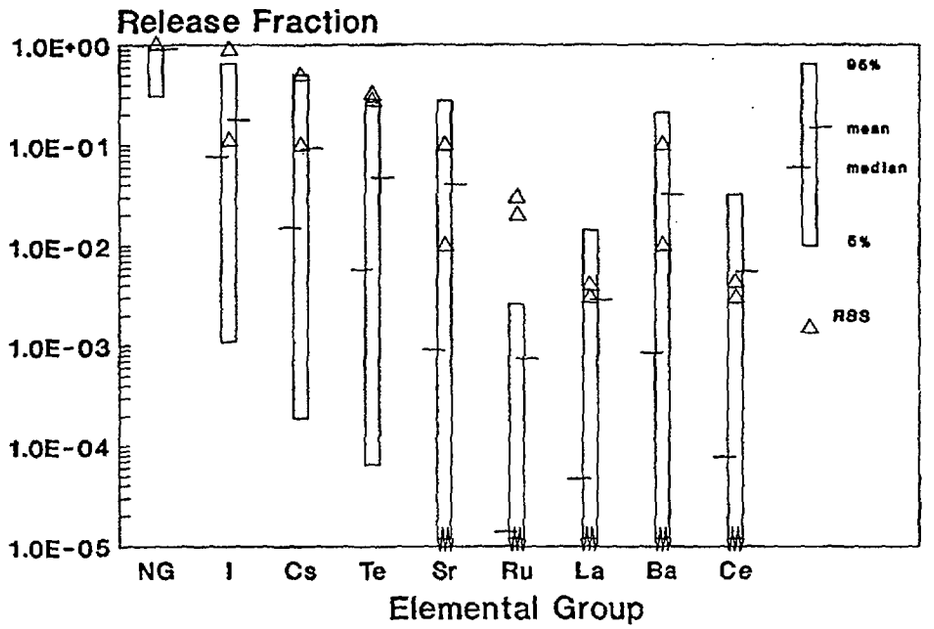
10.4.2 Important Design Features

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

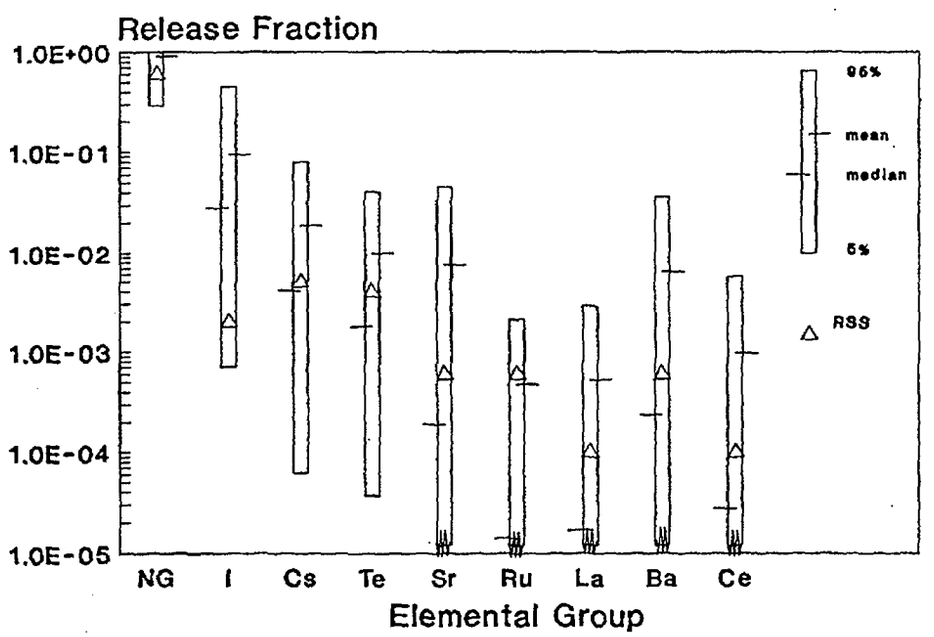
Suppression Pools

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

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



a. Comparison with Bins BWR2 and BWR3



b. Comparison with Bin BWR4

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

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

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

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

Containment Sprays

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

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

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

Ice Condenser

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

Drywell-Wetwell Configuration

The Mark III design has the apparent advantage, relative to the Mark I and Mark II designs, of the wetwell boundary completely enclosing the drywell, in effect providing a double barrier to radioactive material release. As long as the drywell remains intact, any release of radioactive material from the fuel would be subject to decontamination by the suppression pool. For this reason, failure

of the Mark III containment is not as important to severe accident risk as the potential for containment failure in combination with drywell failure. Figures 6.5 and 6.6 illustrate the difference in the environmental source terms for the early containment failure bins with and without drywell failure. With the drywell intact, the environmental source term is reduced to a level at which early fatalities would not be expected to occur, even for early failure of the outer containment. The potential advantages of the drywell-wetwell configuration were found to be limited in this study by the significant probability of drywell failure in an accident.

Cavity Flooding

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

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

Reactor Building/Auxiliary Building Retention

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

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

Containment Venting

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

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

10.4.3 Important Phenomenological Uncertainties

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

the importance of uncertain variables as a function of the magnitude of the environmental release.

Relative Importance of Source Term Variables

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

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

Importance of Source Term Variables to Uncertainty in Environmental Release

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

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

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

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

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

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

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

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

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

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

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

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

11. PERSPECTIVES ON OFFSITE CONSEQUENCES

11.1 Introduction

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

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

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

11.2 Discussion of Consequence CCDFs

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

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

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

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

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

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

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

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

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

Early Fatality Magnitudes

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Table 11.3 Offsite protective measures assumptions.

1. **Emergency Response Assumptions**

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

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

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

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

   b. Outside of 10-mile EPZ:

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

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

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

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

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

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

      from groundshine and inhalation of resuspended radionuclides.

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

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

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

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

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

Latent Cancer Fatality Magnitudes

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

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

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

Population Exposure Magnitudes

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

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

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

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

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

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

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

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

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

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

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

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

- Protective actions are not assumed for water ingestion.

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

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

11.4 Comparison with Reactor Safety Study

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

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

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

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

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

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

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

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

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

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

11.5 Uncertainties and Sensitivities

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

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

11.6 Sensitivity of Consequence Measure CCDFs to Protective Measure Assumptions

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

11.6.1 Sensitivity of Early Fatality CCDFs to Emergency Response

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

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

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

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

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

Zion

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

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

Sequoyah

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

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

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

Peach Bottom, Grand Gulf

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

Surry

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

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

a. Evacuation (see Table 11.3).

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

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

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

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

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

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

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

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

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

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

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

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

The potential for latent cancer fatalities and population exposure is assumed to exist down to any low level of radiation dose and, therefore, over the entire site region. Although both early and chronic exposure pathways contribute to these consequence measures, only the chronic exposure pathways are expected to be mitigated by the long-term countermeasures such as decontamination, temporary interdiction, or condemnation of contaminated land, property, and foods based on guidance provided by responsible Federal agencies in terms of PAGs. This implies that, if the radiation dose to an individual from a

chronic exposure pathway would be projected to exceed the PAG (or intervention) level for that pathway, countermeasures should be undertaken to reduce the projected dose from the pathway so that it does not exceed the PAG level. Therefore, the latent cancer fatalities and the population exposures stemming from the chronic exposure pathways are expected to be sensitive to the PAG values.

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

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

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

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

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

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

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

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

12. PERSPECTIVES ON PUBLIC RISK

12.1 Introduction

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

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

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

12.2 Summary of Results

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

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

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

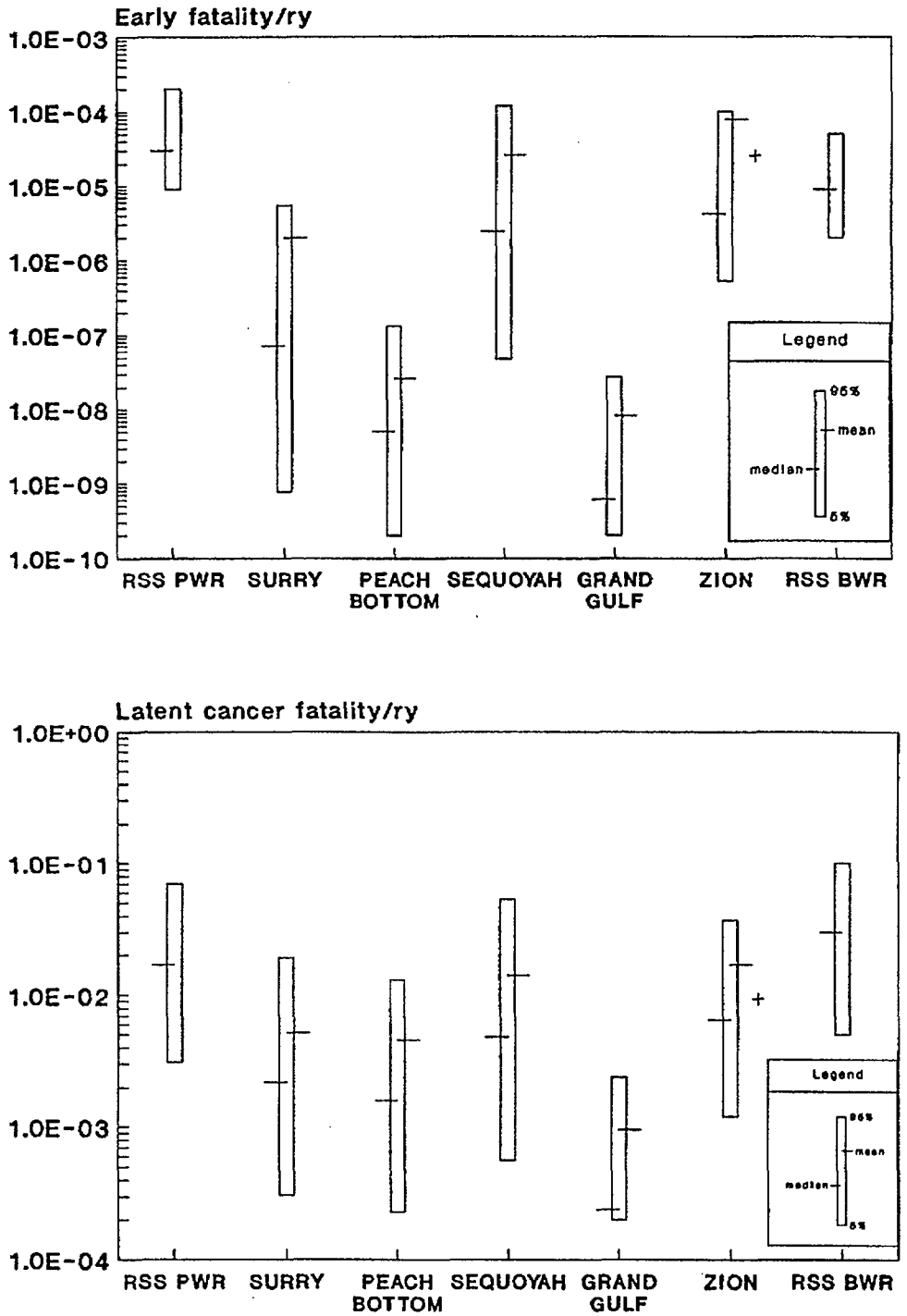
- The risks to the public from operation of the five plants are, in general, lower than the Reactor Safety Study (Ref. 12.10) estimates for two plants in 1975. Among the five plants studied, the two BWRs show lower risks than the three PWRs, principally because of the much lower core damage frequencies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

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

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

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

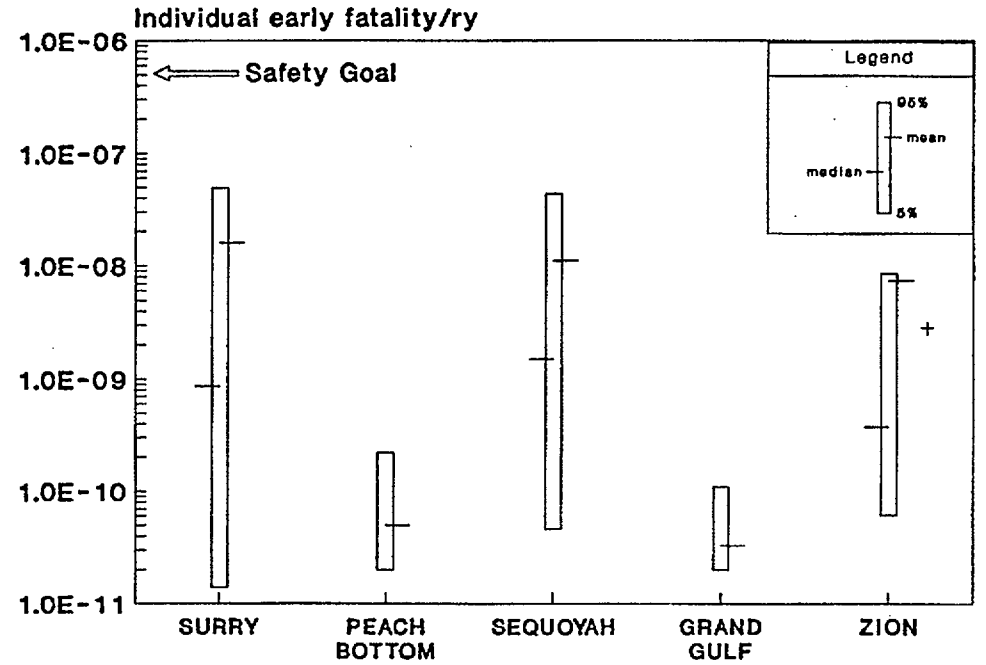
*Throughout this report, discussion of and comparison with the NRC safety goals relates specifically and only to the two quantitative health objectives identified in the Commission's policy statement (Ref. 12.8).

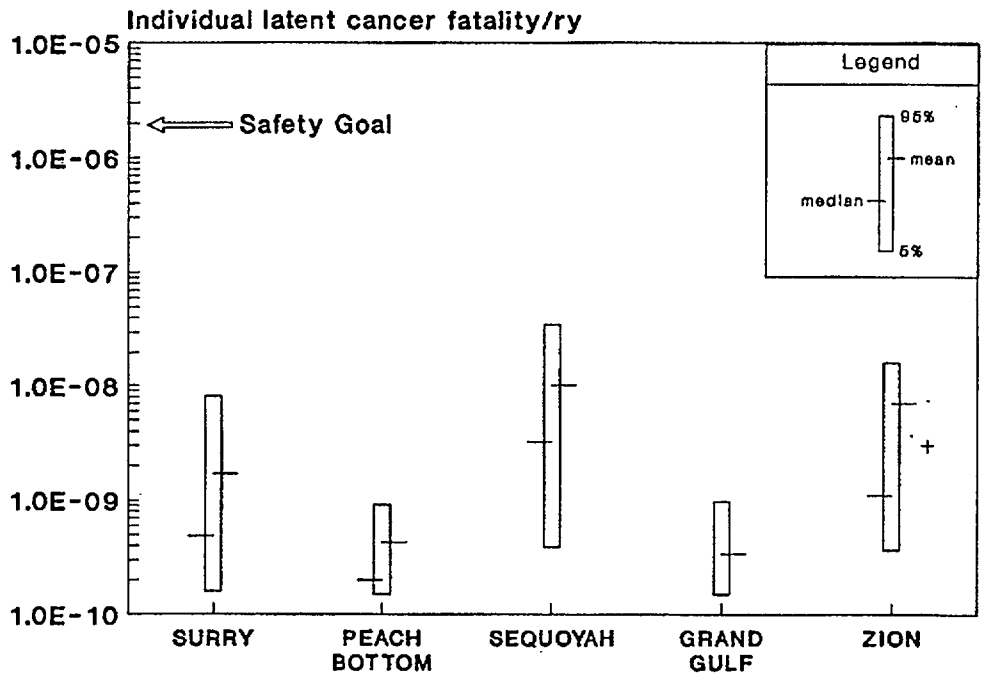


Notes: As discussed in Reference 12.9, estimated risks at or below 1E-7 should be viewed with caution because of the potential impact of events not studied in the risk analyses.

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

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

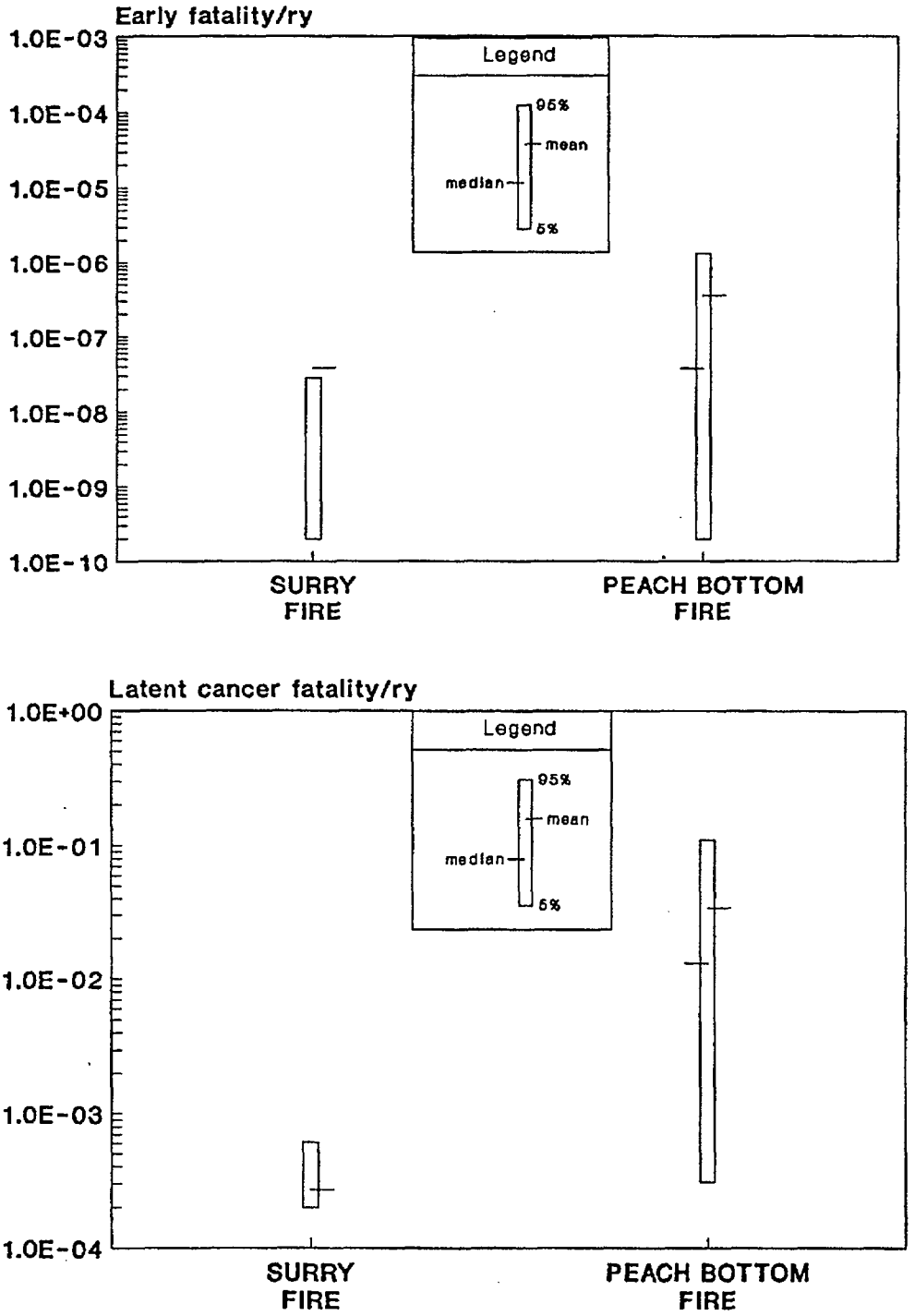




Notes: As discussed in Reference 12.9, estimated risks at or below 1E-7 should be viewed with caution because of the potential impact of events not studied in the risk analyses.

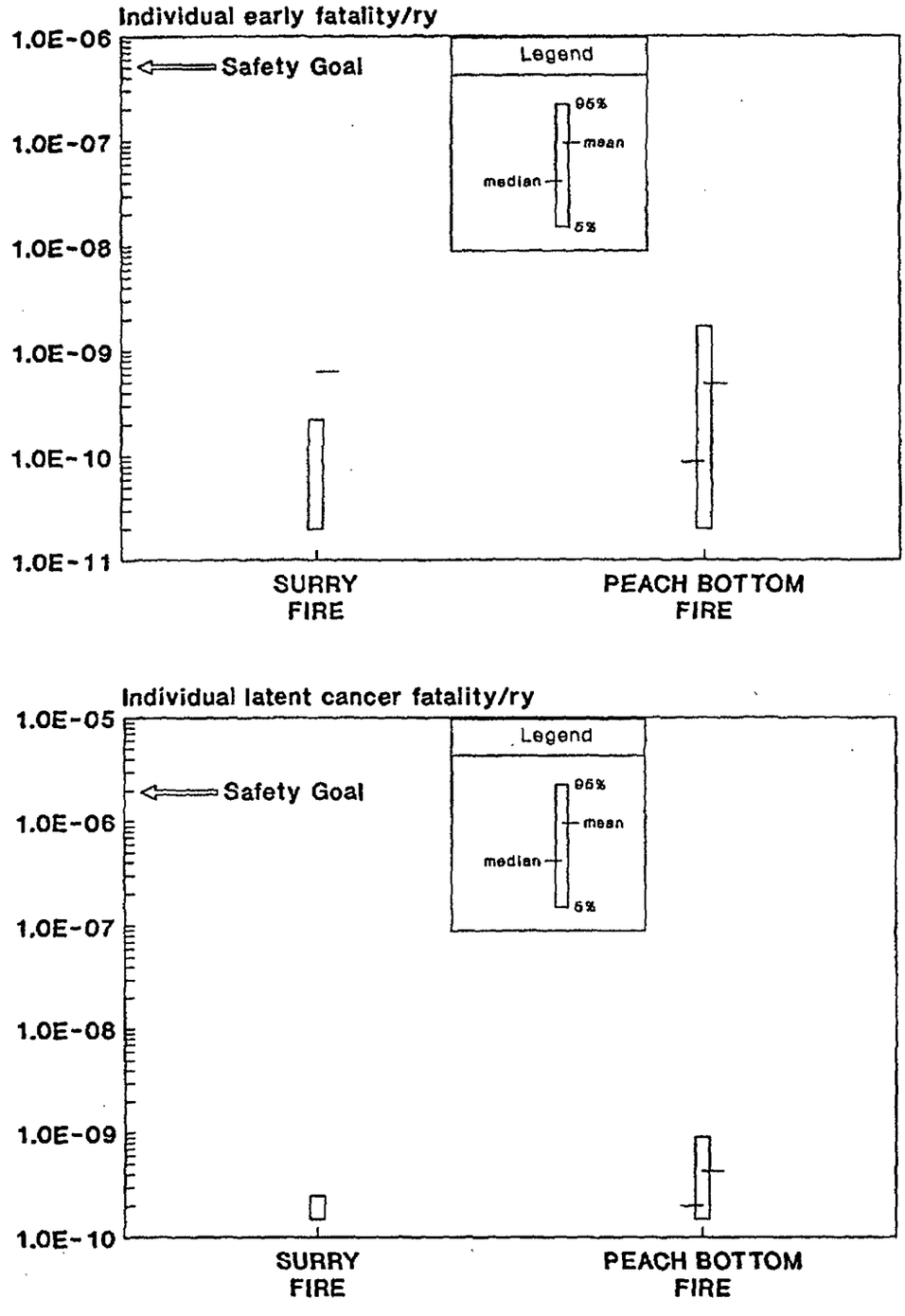
"+" indicates recalculated mean value based on recent modifications to the Zion plant (as discussed in Section C.15).

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



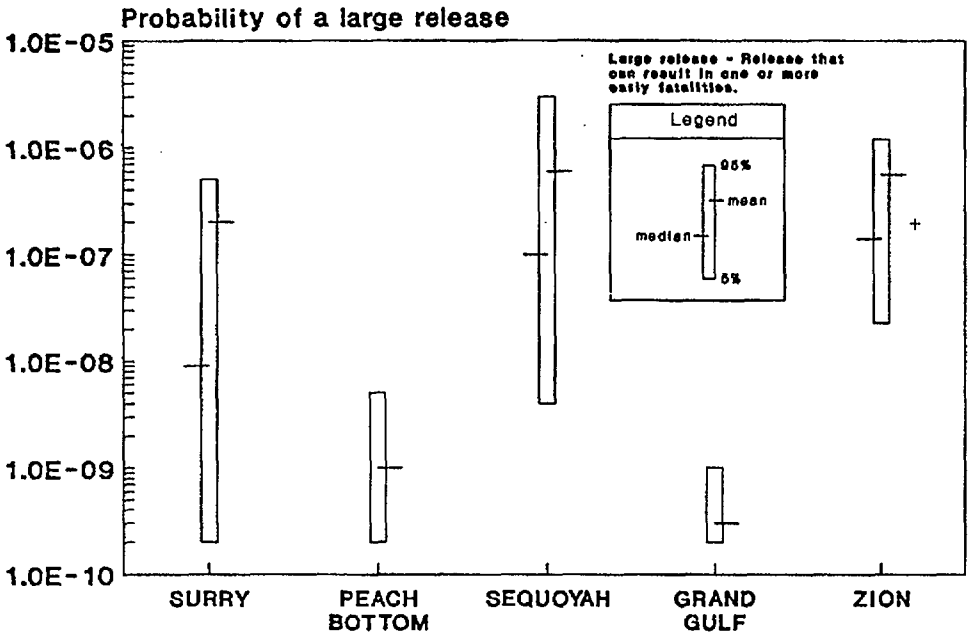
Note: As discussed in Reference 12.9, estimated risks at or below 1E-7 should be viewed with caution because of the potential impact of events not studied in the risk analyses.

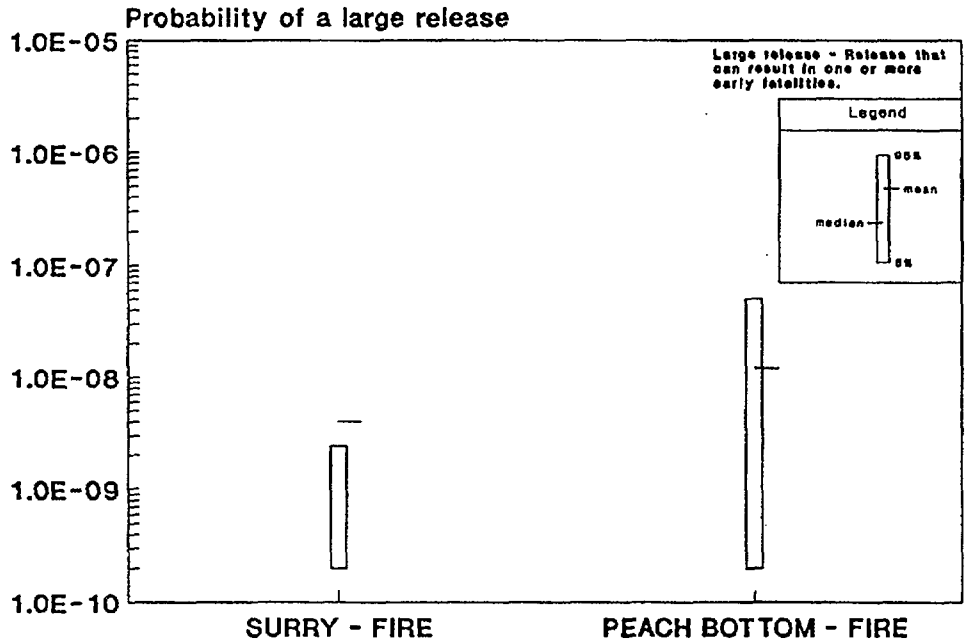
Figure 12.3 Comparison of early and latent cancer fatality risks at Surry and Peach Bottom (fire-initiated accidents).



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

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





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

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

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

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

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

12.3 Comparison with Reactor Safety Study

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

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

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

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

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

12.4 Perspectives

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

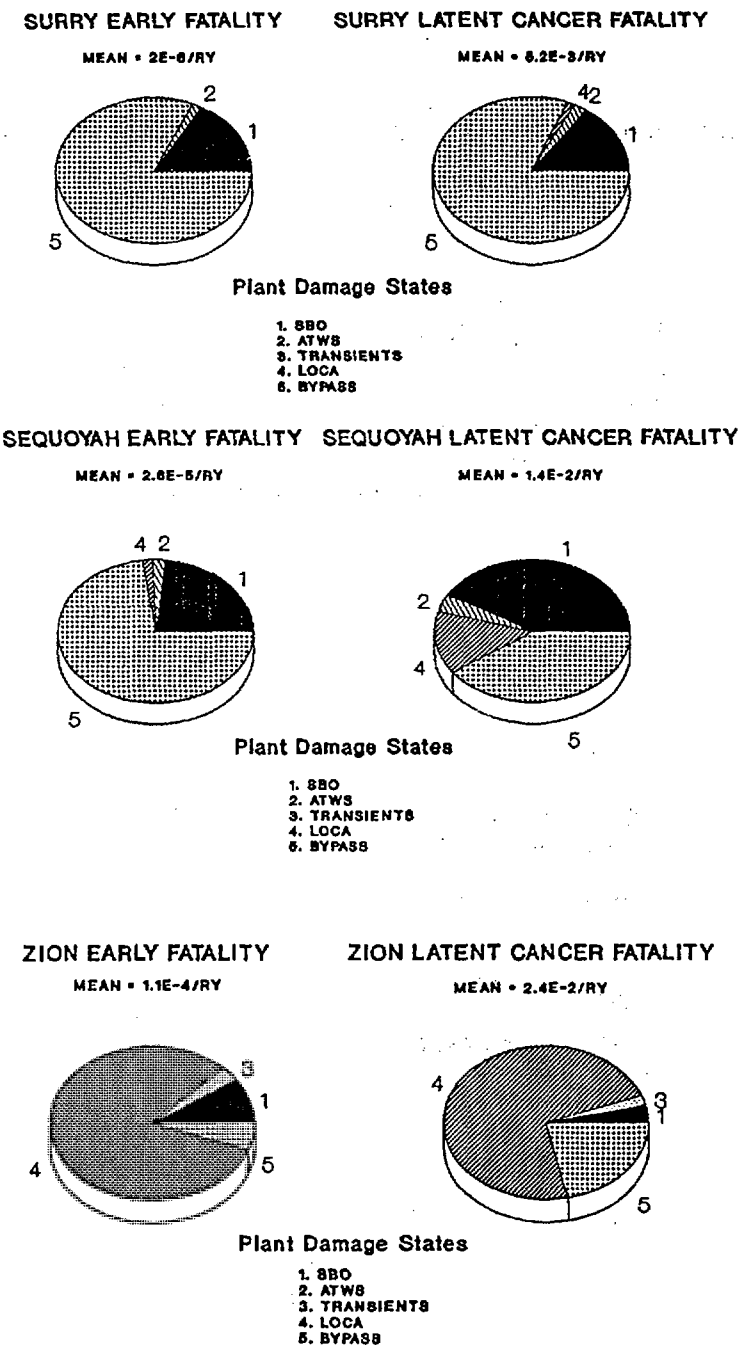


Figure 12.6 Contributions of plant damage states to mean early and latent cancer fatality risks for Surry, Sequoyah, and Zion (internal events).

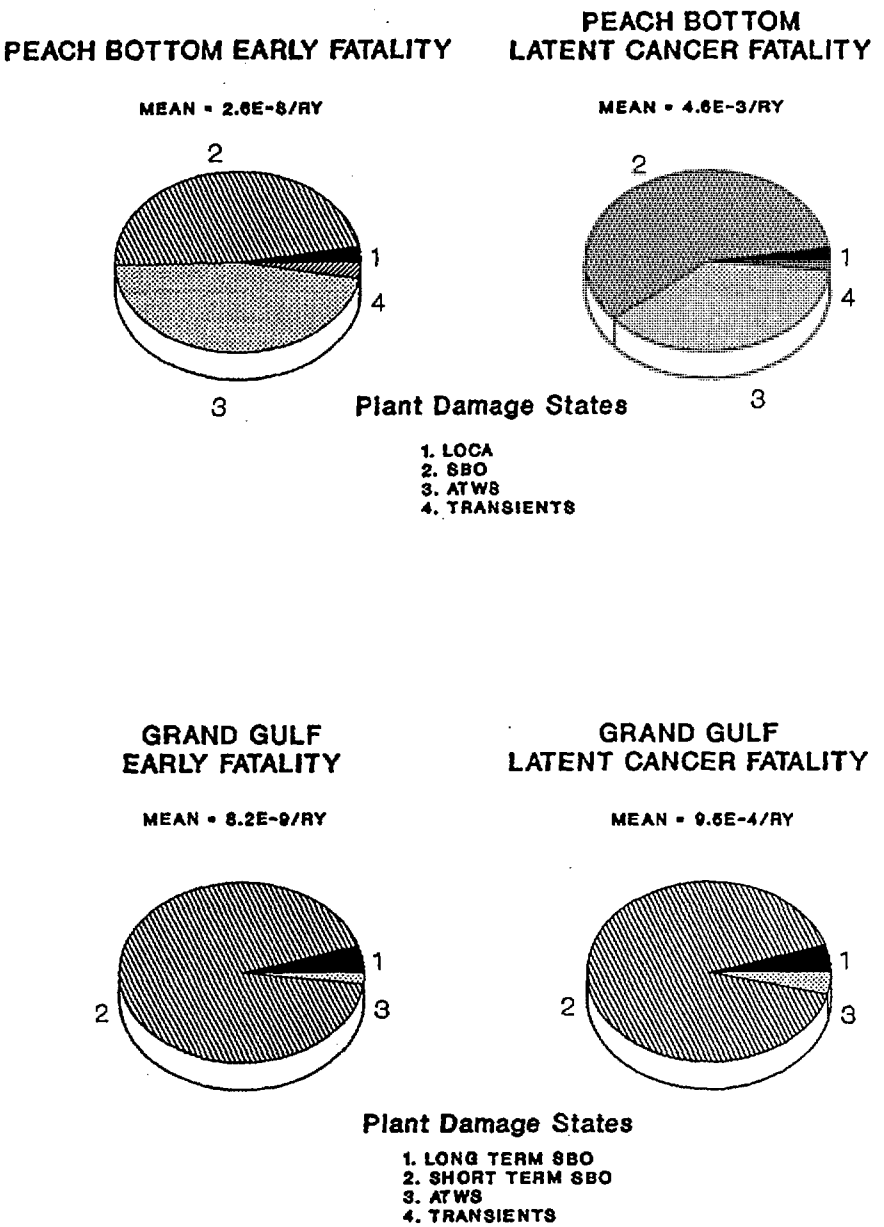


Figure 12.7 Contributions of plant damage states to mean early and latent cancer fatality risks for Peach Bottom and Grand Gulf (internal events).

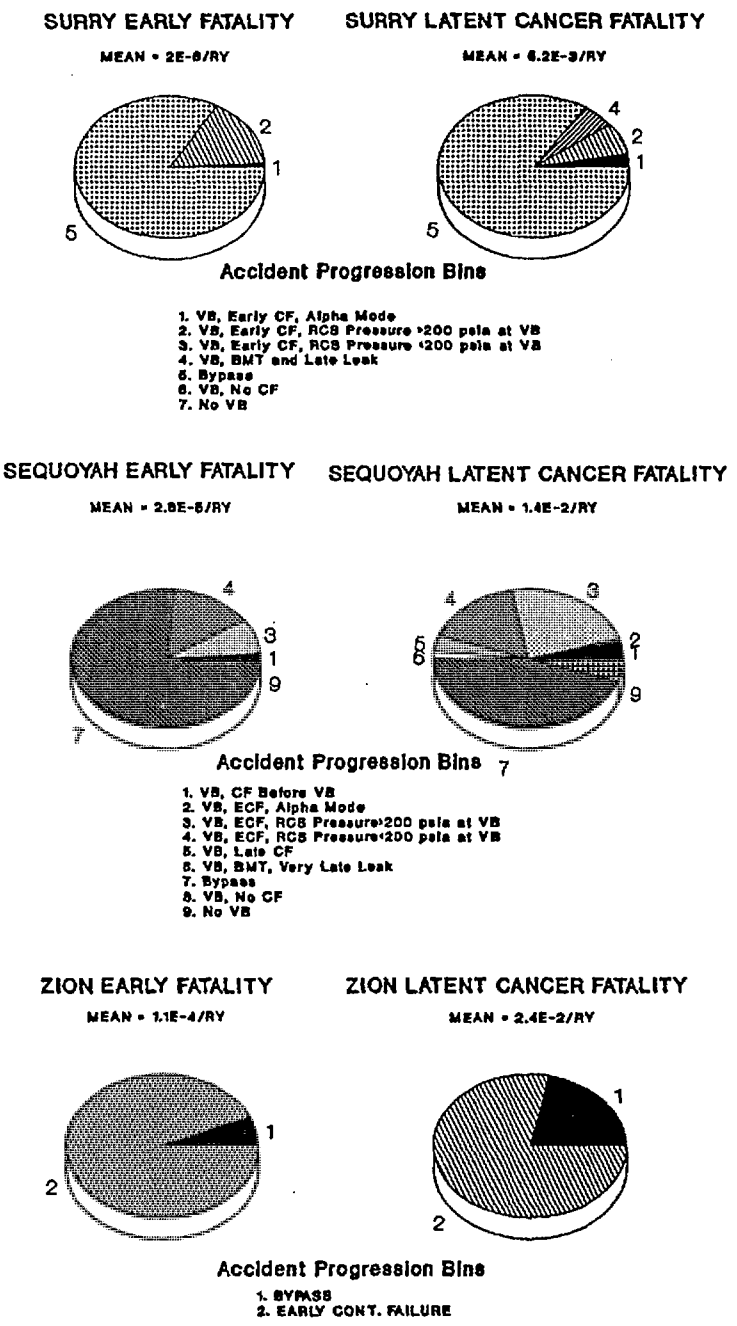


Figure 12.8 Contributions of accident progression bins to mean early and latent cancer fatality risks for Surry, Sequoyah, and Zion (internal events).

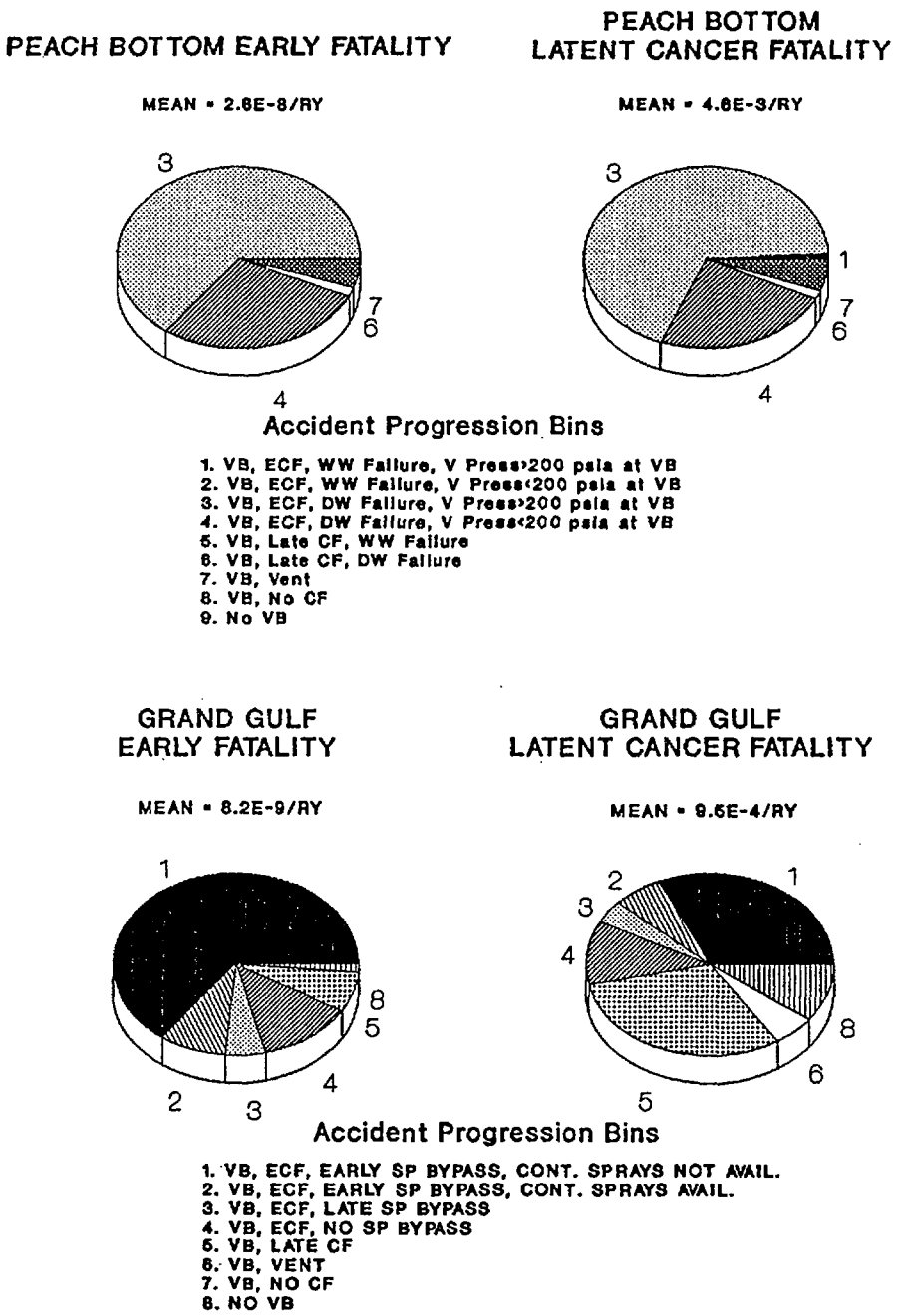


Figure 12.9 Contributions of accident progression bins to mean early and latent cancer fatality risks for Peach Bottom and Grand Gulf (internal events).

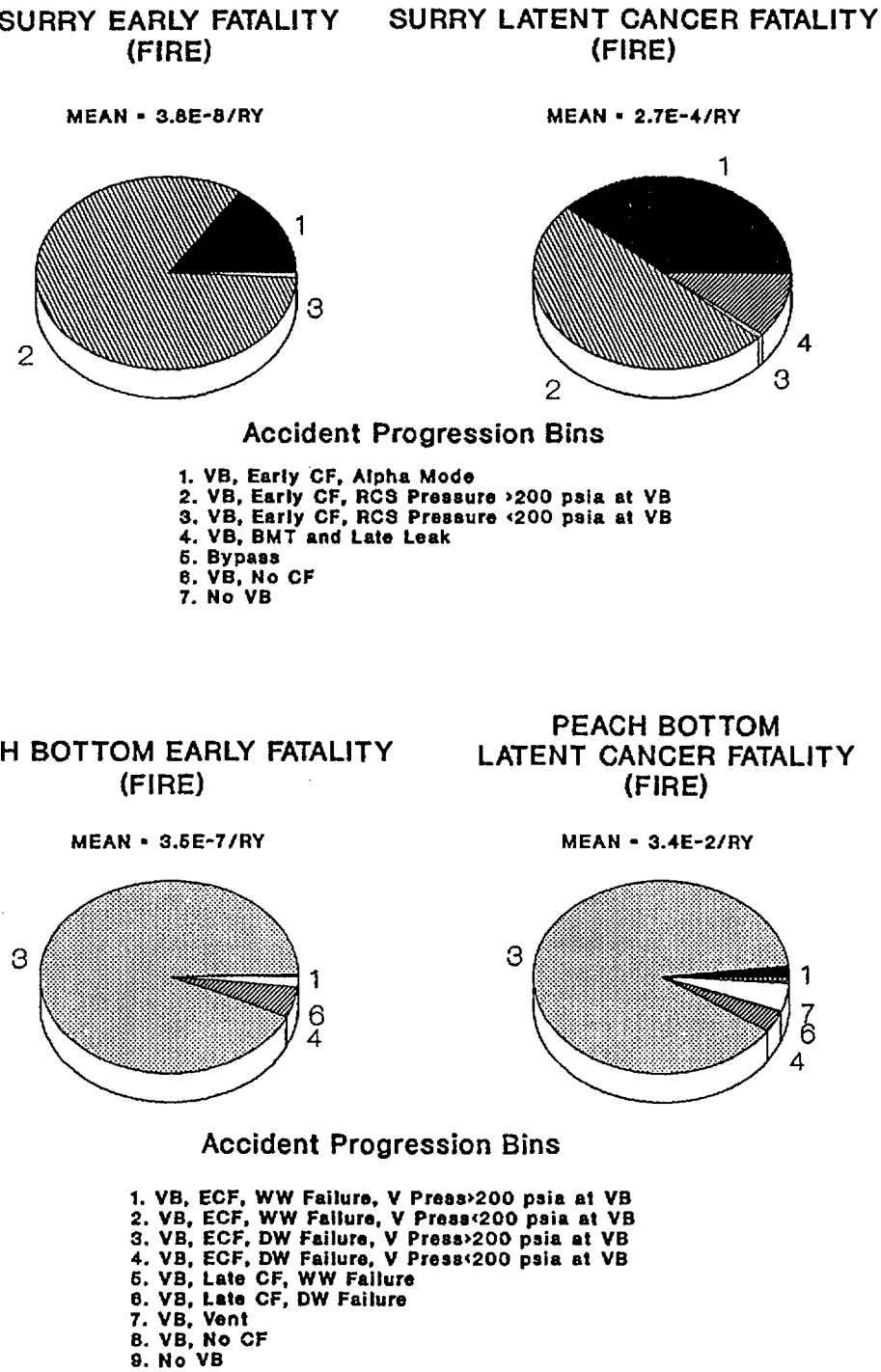


Figure 12.10 Contributions of accident progression bins to mean early and latent cancer fatality risks for Surry and Peach Bottom (fire-initiated accidents).

Accident Sequences Important to Risk

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

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

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

Containment Failure Issues Important to Risk

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

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

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

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

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

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

- At Sequoyah, containment bypass events are assessed to be most important to mean early fatality risk. Another failure important to

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

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

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

Source Term and Offsite Consequence Issues Important to Risk

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

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

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

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

Factors Important to Uncertainty in Risk

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

The results of the regression analyses indicate the following:

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

while other parameters such as those relating to source terms in containment, containment strength, etc., are not found to be important.

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

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

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

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

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

accident frequency parameters (fire initiating event frequencies, diesel generator failure-to-run probability).

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

Impact of Emergency Response and Protective Action Guide Options

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

Emergency Response Options

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

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

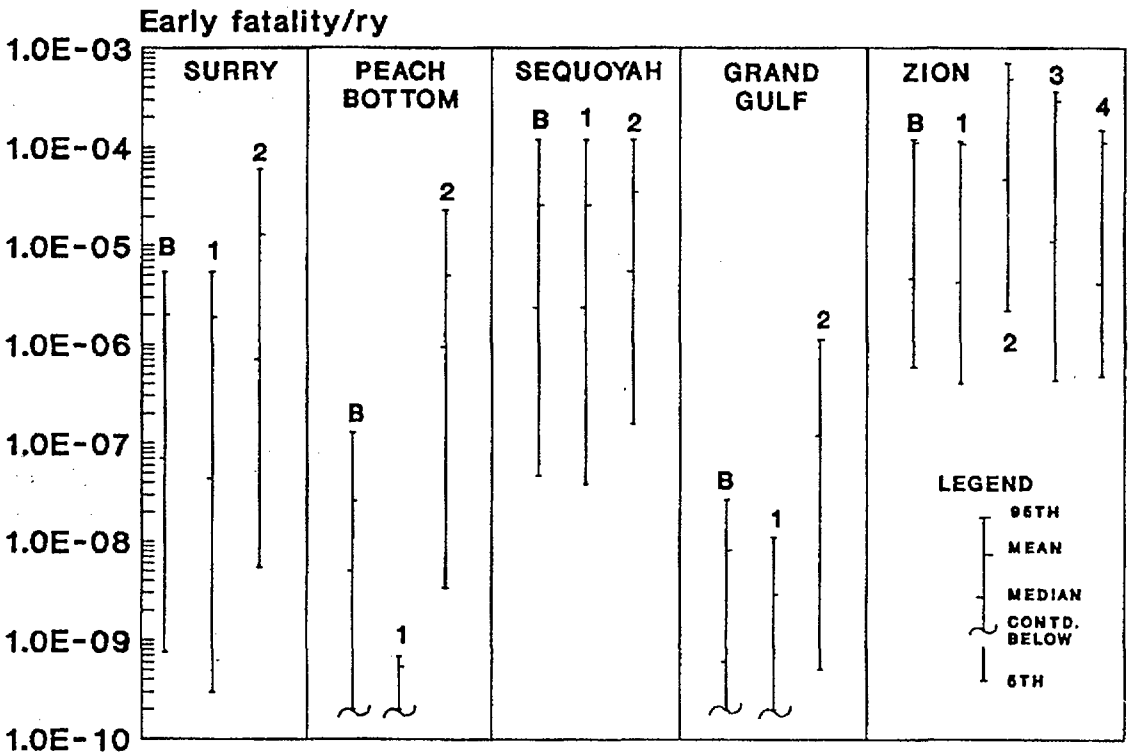
The last two options are used in the Zion plant analysis only. Results of the analyses are presented in Figure 12.11.

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

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

Protective Action Options

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



BASE CASE (B)

99.5% Evacuation from 0 to 10 miles

EMERGENCY RESPONSE OPTIONS (1 TO 4)

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

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

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

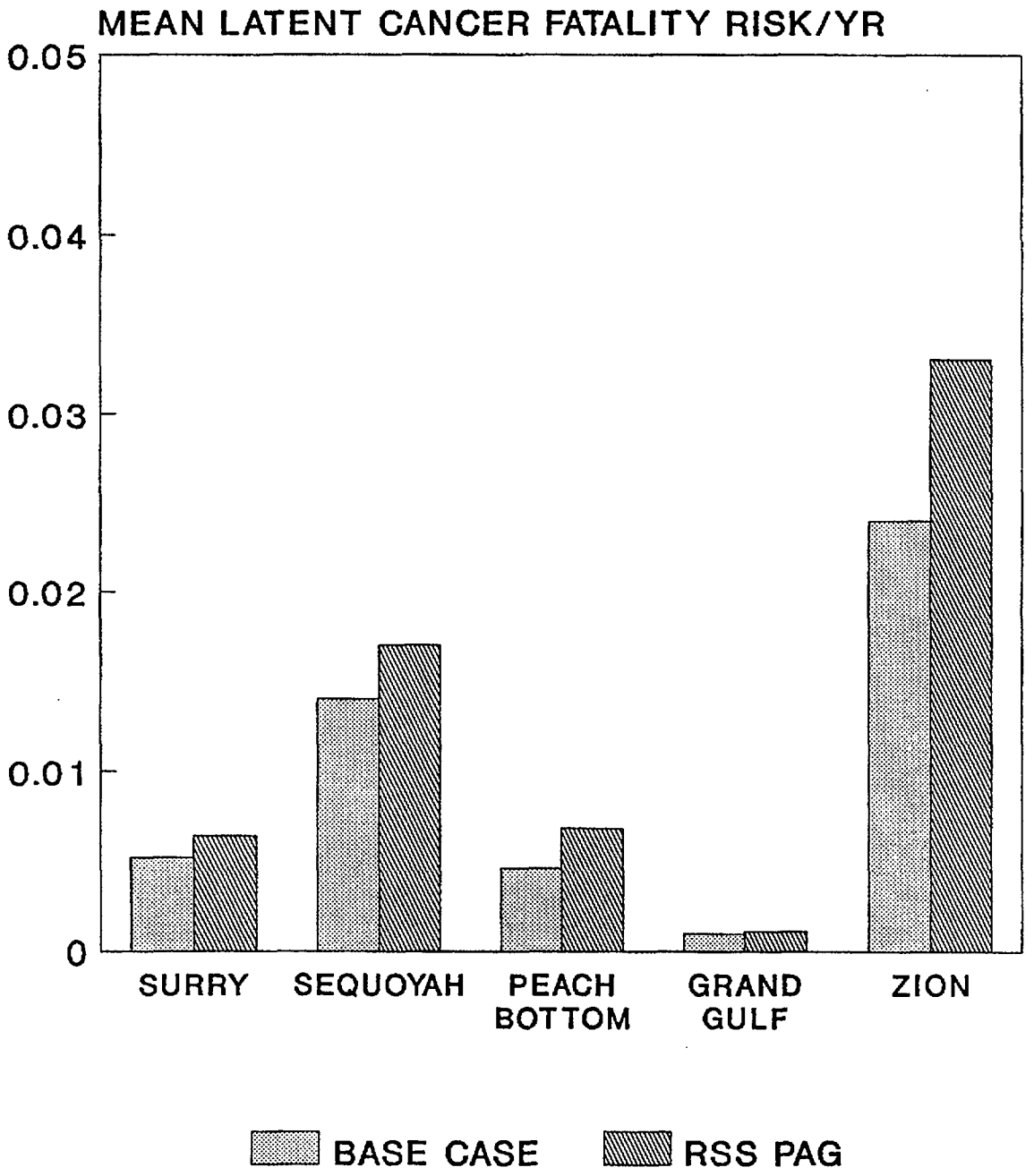


Figure 12.12 Effects of protective action assumptions on mean latent cancer fatality risks at all plants (internal events).

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

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

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

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

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

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

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

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

13. NUREG–1150 AS A RESOURCE DOCUMENT

13.1 Introduction

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

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

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

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

13.2 Probabilistic Models of Accident Sequences

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

13.2.1 Guidance for Individual Plant Examinations

Plant-specific PRAs have yielded valuable perspectives on unique plant vulnerabilities. The NRC and the nuclear industry both have considerable experience with plant-specific PRAs. This experience, coupled with the interactions of NRC and the nuclear industry on severe accident issues, have resulted in the Commission's formulating an integrated systematic approach to an examination of each nuclear power plant now operating or under construction for possible significant risk contributions (sometimes called "outliers") that might be plant specific and might be

missed without a systematic approach. In November 1988, the NRC requested (by generic letter) that each licensed nuclear power plant perform an individual plant examination (IPE) to identify any plant-specific vulnerabilities to severe accidents (Ref. 13.17). The technical data generated in the course of preparing NUREG–1150 on severe accident frequencies, risks, and important uncertainties were used in developing the analysis requirements described in the IPE generic letter and the supplemental guidance on the IPE external-event analysis (Ref. 13.18).* These studies will also aid the staff in evaluating individual submittals, assessing the adequacy of the identification of plant-specific vulnerabilities by the licensee, and evaluating any associated potential plant modifications.

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

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

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

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

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

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

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

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

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

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

13.2.2 Guidance for Accident Management Strategies

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

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

PRA information has been used in the past to influence accident management strategies; however, the methods used in NUREG–1150 can bring added depth and breadth to the process, along with a detailed, explicit treatment of uncertainties. The integrated nature of the methods is particularly important, since actions taken during early parts of an accident can affect later accident progression and offsite consequences. For example, an accident management strategy at a BWR may involve opening a containment vent. This action can affect such things as the system response and core damage frequency, the retention of radioactive material within the containment, and the timing of radionuclide releases (which impacts evacuation strategies). It is possible that actions to reduce the core damage frequency can yield accident sequences of lower frequency but with much higher consequences. All these factors need to be considered in concert when developing

appropriate venting strategies. The treatment of uncertainties is another key aspect of accident management. Generally, procedures are developed based on "most likely" or "expected" outcomes. For severe accidents, the outcomes are particularly uncertain. PRA models and results, such as those produced in the accident progression event trees, can identify possible alternative outcomes for important accident sequences. By making this information available to operators and response teams, unexpected events can be recognized when they occur, and a more flexible approach to severe accidents can be developed. The recent trend toward symptom-based, as opposed to event-based, procedures is consistent with this need for flexibility.

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

Effect of Firewater System at Grand Gulf

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

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

Effect of Feed and Bleed on Core Damage Frequency at Surry

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

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

Gas Turbine Generator Recovery Action at Surry

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

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

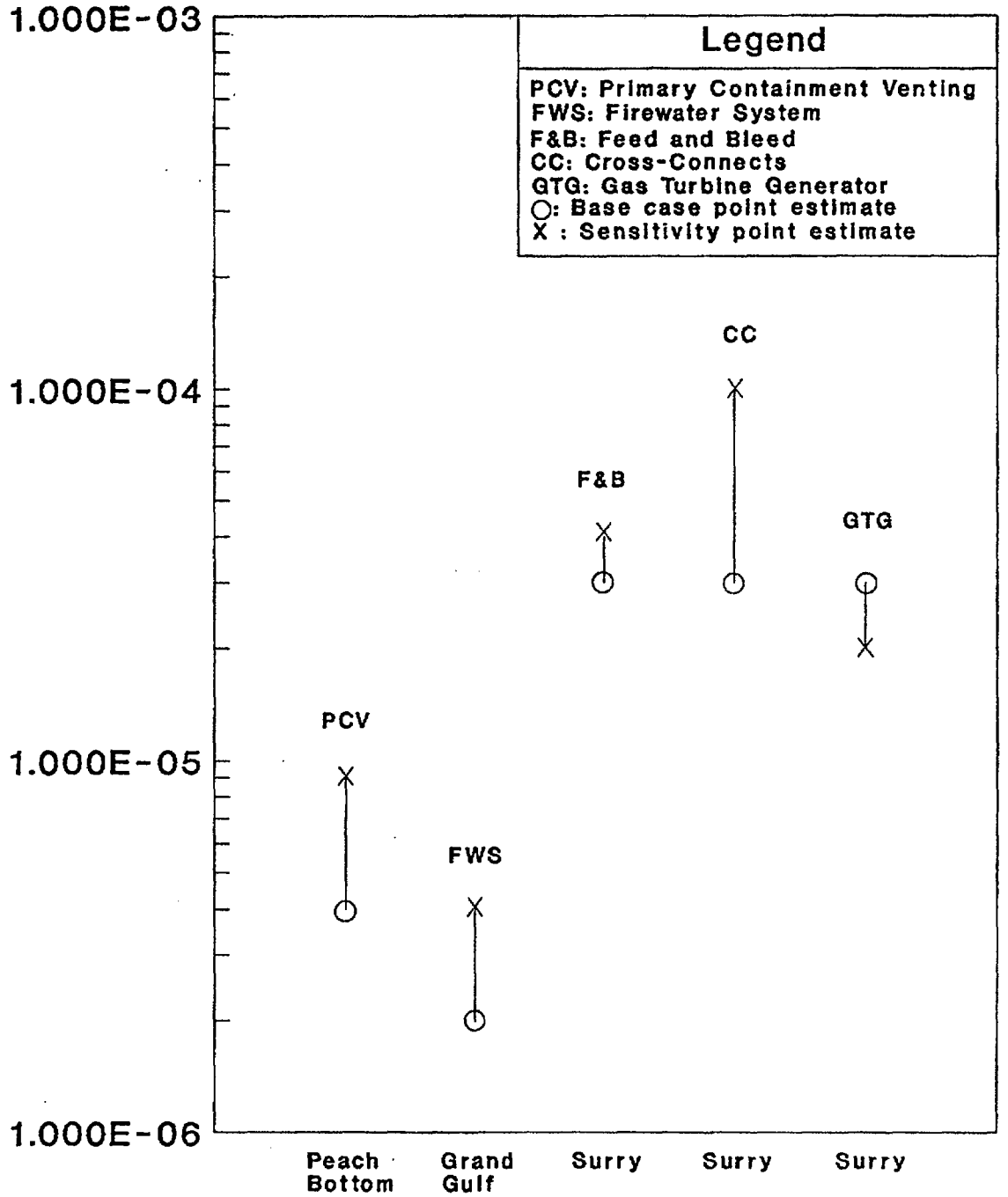


Figure 13.1 Benefits of accident management strategies.

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

High-Pressure Injection and Auxiliary Feedwater Crossconnects at Surry

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

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

Primary Containment Venting at Peach Bottom

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

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

13.2.3 Improving Containment Performance

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

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

13.2.4 Determining Important Plant Operational Features

NUREG-1150 will provide a source of information for investigating the importance of operational safety issues that may arise during day-to-day plant operations. The NUREG-1150 models, methods, and results have already been used to analyze the importance of venting of the suppression pool, the importance of keeping the PORVs and atmospheric dump valves unblocked, the importance of operational characteristics of the ice condenser containment design, the importance of operator recovery during an accident sequence, and the importance of crossties between systems. These operational and system characteristics, as well as many others, are described in detail in Chapters 3 through 7. For example, characteristics of the Surry plant design and operation that

have been found to be important include crossties between units, diesel generators, reactor coolant pump seals, battery capacity, capability for feed and bleed core cooling, subatmospheric containment operation, post-accident heat removal system, and reactor cavity design.

13.2.5 Alternative Safety Goal Implementation Strategies

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

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

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

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

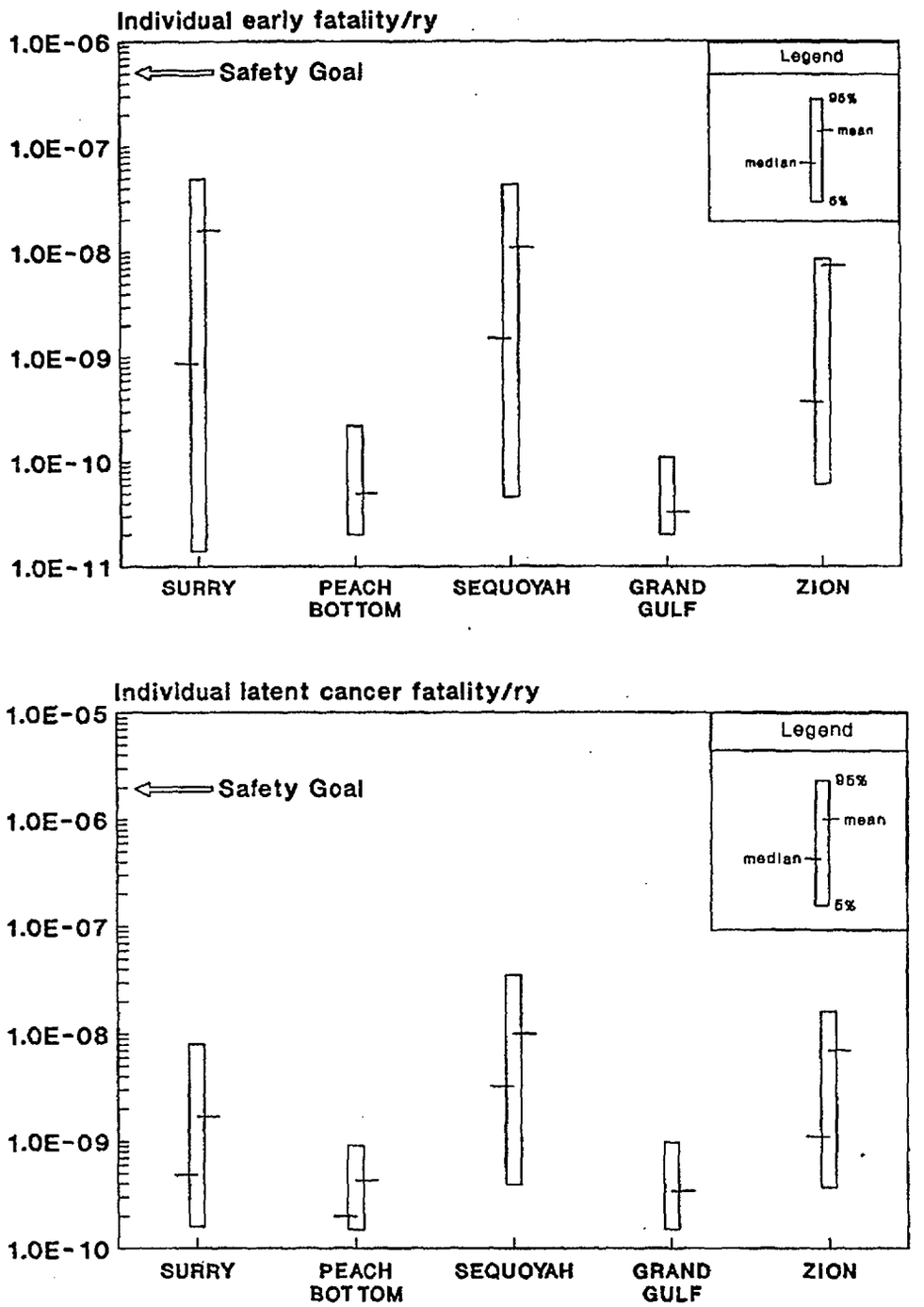
The NUREG–1150 results can also be used to support the analysis of alternative safety goal implementation approaches. One subject of discussion in the staff's work is the need for a supplemental definition of containment performance in severe accidents using the probability of a large release as a measure.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

13.2.6 Effect of Emergency Preparedness on Consequence Estimates

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

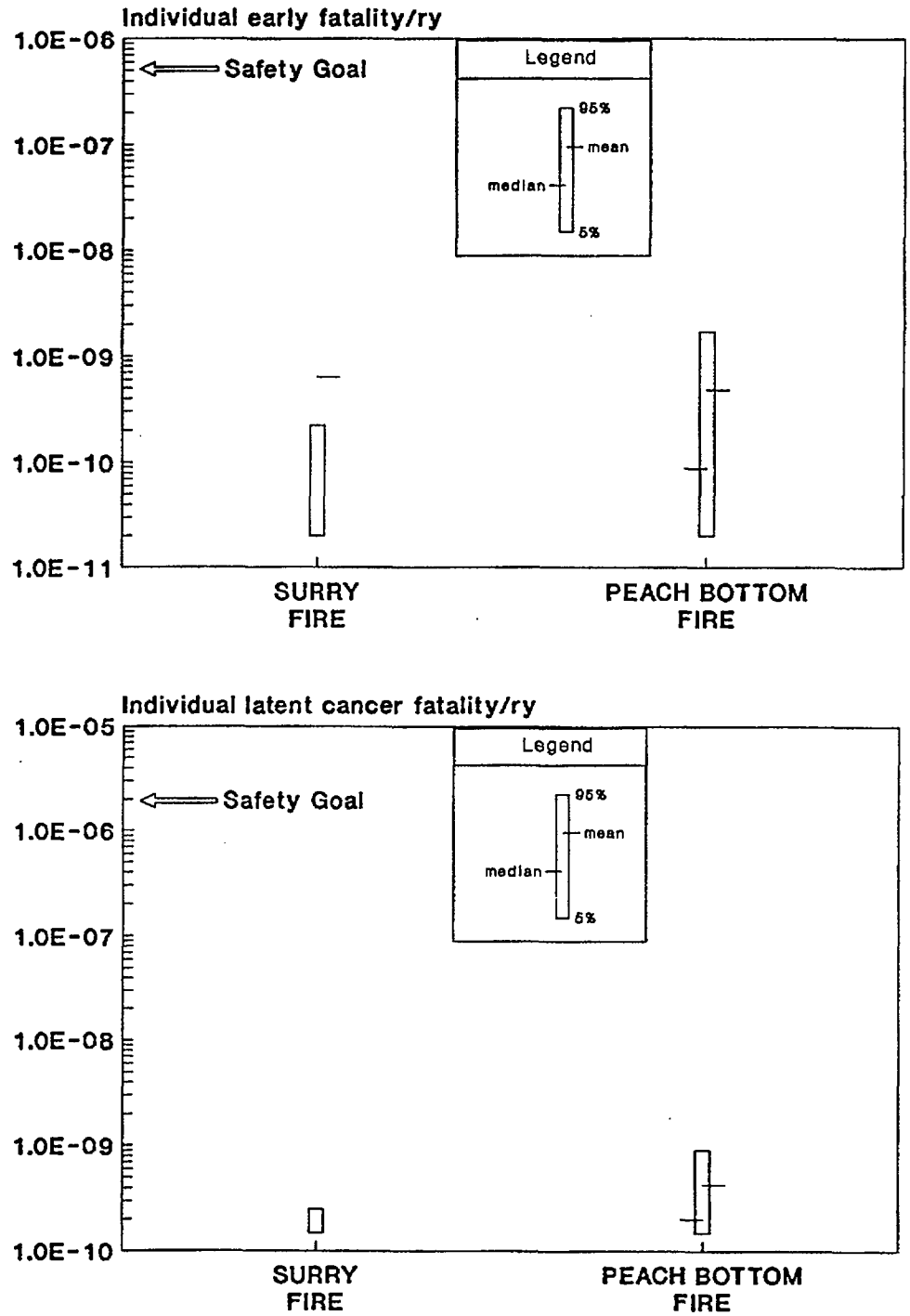
NUREG–1150 provides assessments of a broad spectrum of potential core damage accidents that could occur at a nuclear power plant. These assessments permit the evaluation of hypothetical

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



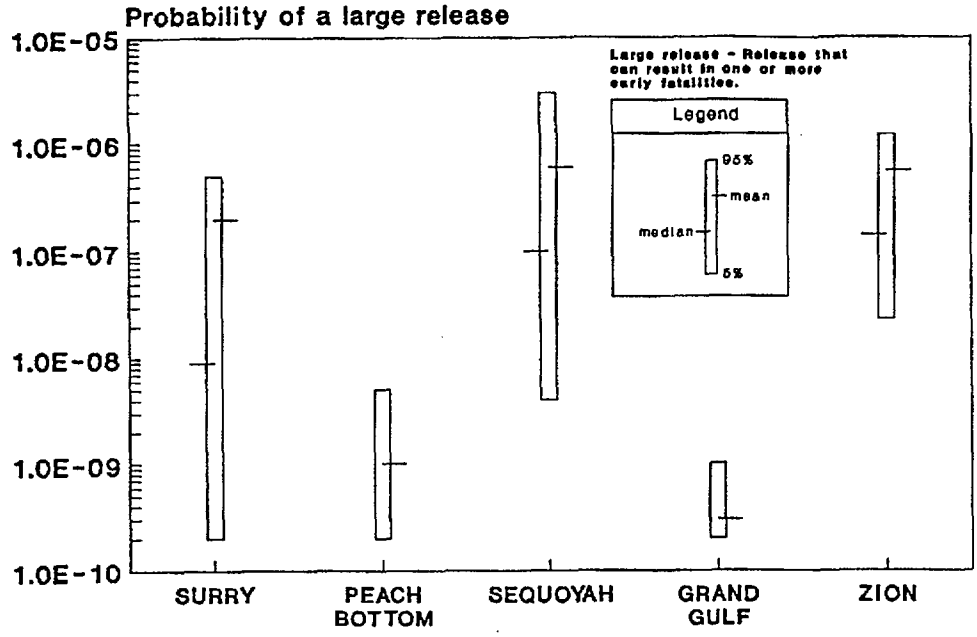
Note: As discussed in Reference 13.23, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

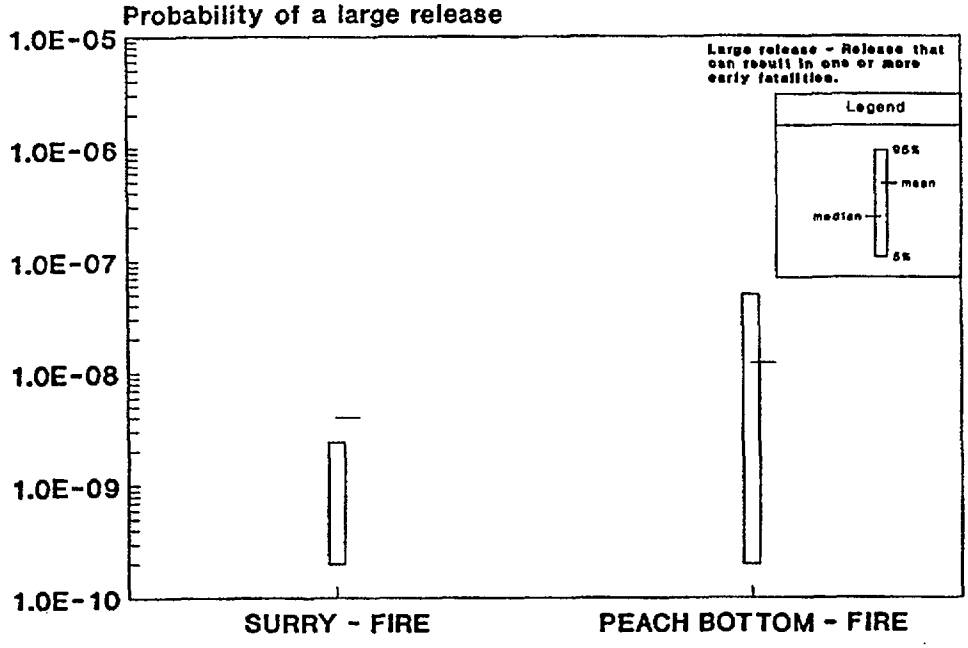
Figure 13.2 Comparison of individual early and latent cancer fatality risks at all plants (internal initiators).



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

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





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

Figure 13.4 Frequency of one or more early fatalities.

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

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

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

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

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

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

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

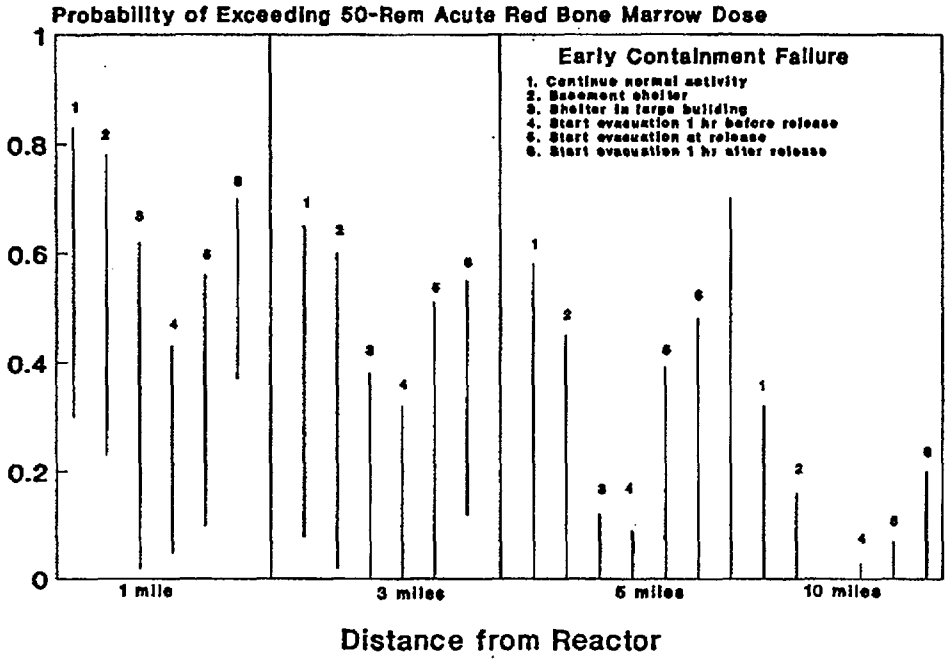
At 10 miles, no protective actions except relocation would be necessary to avoid 200-rem doses. Sheltering in large buildings or evacuation prior to release would probably keep doses below 50 rems.

13.3 Major Factors Contributing to Risk

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

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

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



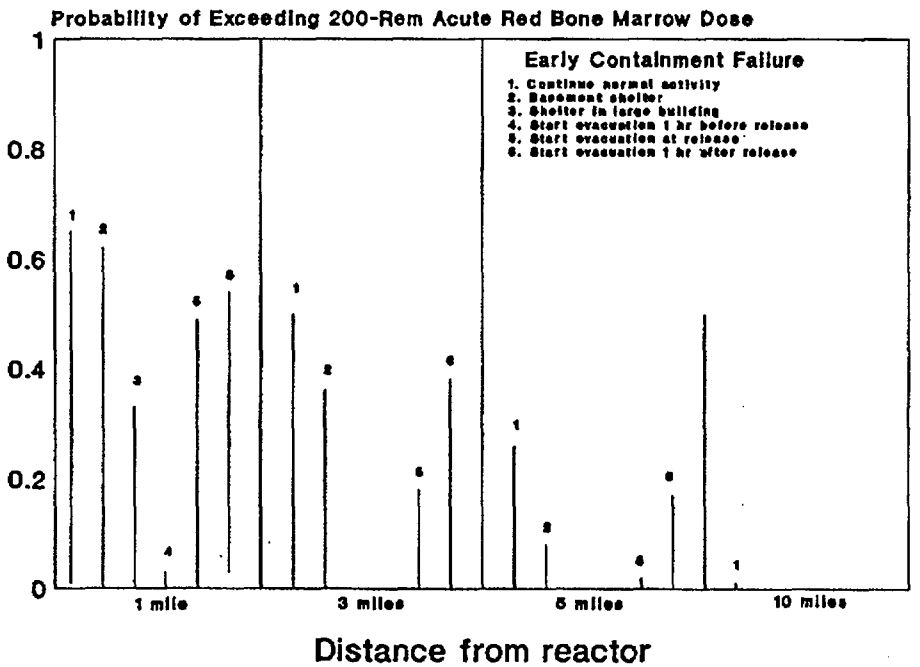
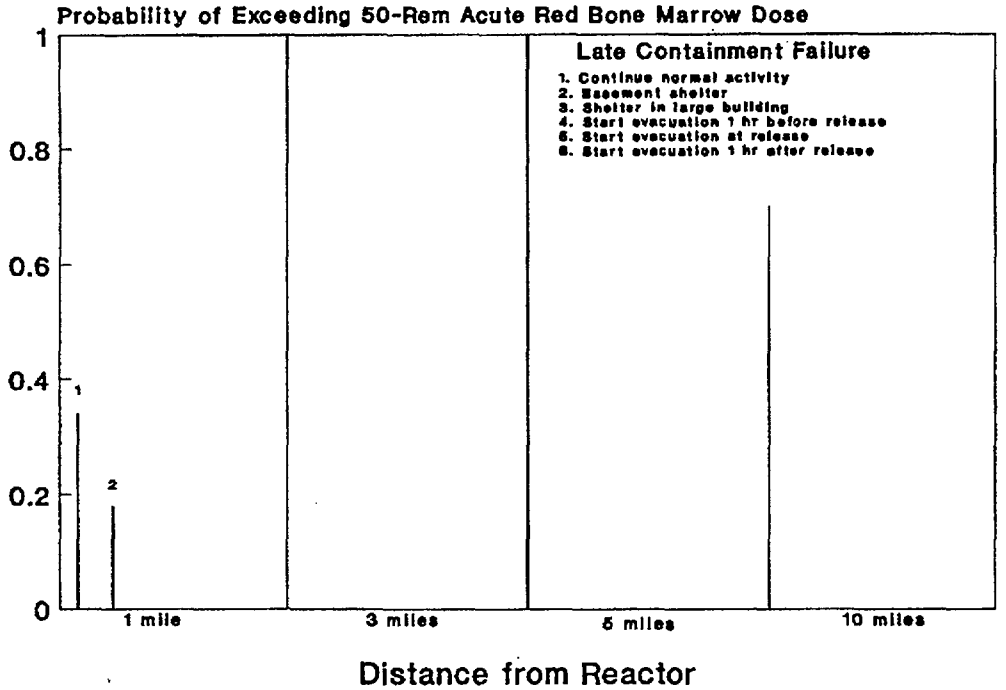


Figure 13.5 Relative effectiveness of emergency response actions assuming early containment failure with high and low source terms.



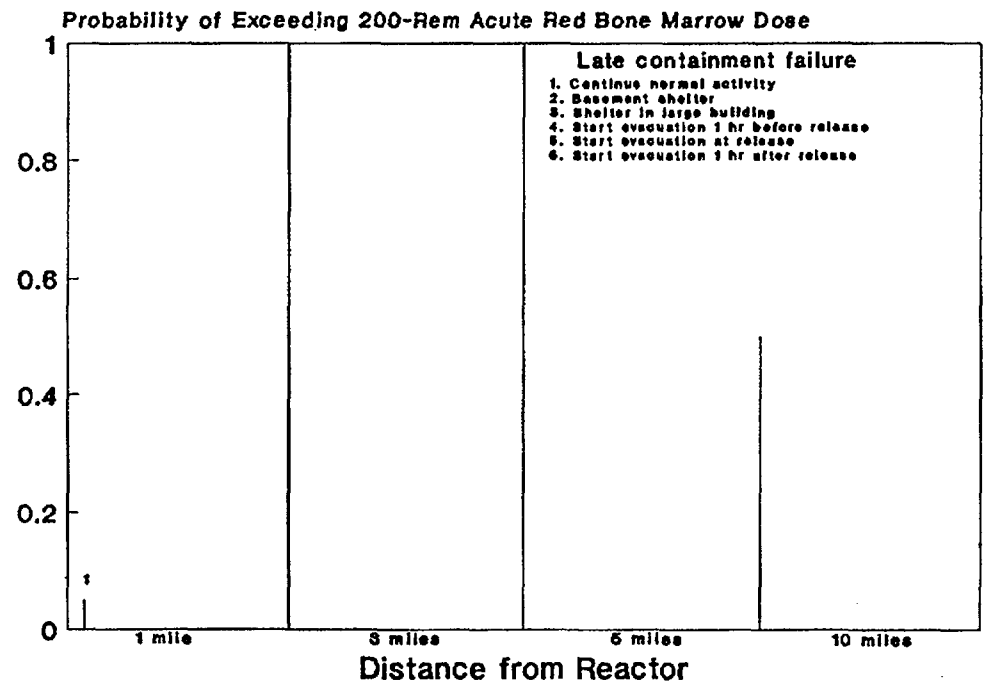


Figure 13.6 Relative effectiveness of emergency response actions assuming late containment failure with high and low source terms.

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

13.3.1 Reactor Research

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

13.3.2 Prioritization of Generic Issues

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

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

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

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

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

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

13.3.3 Use of PRA in Inspections

The importance to NRC of risk-based inspection data is exemplified by the following statement in NRC's 5-Year Plan: "Probabilistic risk assessment techniques will be applied to all phases of the inspection program in order to insure that inspection activities are prioritized and conducted in an integrated fashion." Within NRC, the Risk Applications Branch of the Office of Nuclear Reactor Regulation has the responsibility of directly

providing risk-based information to the regional offices and resident inspectors. This ongoing effort has resulted in the development of plant-specific, and in some cases generic, PRA perspectives that help to provide an optimization of inspection resources and a prioritization of inspection resources on the high-risk aspects of a plant. Using draft NUREG–1150 data, team inspection procedures based on plant-specific PRA information have been developed and implemented on such plants as Grand Gulf. Formalization of these inspection activities can be found in a recently issued inspection module entitled "Risk Focused Operation Readiness Inspection Procedures." This module focuses on how to use PRA perspectives and conduct a risk-based team inspection based on risk insights. The spectrum of reactor plant design types addressed in NUREG–1150 provide a broad risk data base that in many instances can be used to assist in inspection-type decisions even for plants without a PRA.

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

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

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Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants

Appendices D and E

Final Report

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This report was first published in February 1987 as a draft for public comment. Extensive peer review and public comment were received. As a result, both the underlying technical analyses and the report itself were substantially changed. A second version of the report was published in June 1989 as a draft for peer review. Two peer reviews of the second version were performed. One was sponsored by NRC; its results are published as the NRC report NUREG–1420. A second was sponsored by the American Nuclear Society (ANS); its report has also been completed and is available from the ANS. The comments by both groups were generally positive and recommended that a final version of the report be published as soon as practical and without performing any major reanalysis. With this direction, the NRC proceeded to generate this final version of the report.

Volume 3 of this report contains two appendices. Appendix D summarizes comments received, and staff responses, on the first (February 1987) draft of NUREG–1150. Appendix E provides a similar summary of comments and responses, but for the second (June 1989) version of the report.

lowering the frequency of a legitimate core melt accident sequence by three to four orders of magnitude. There is no reason to believe a value of 9E–9/reactor year when detailed plant-specific PRAs have consistently estimated the frequency in the range of 1.6E–6 to 1.0E–5/reactor year.

Response:

The analysis of steam generator tube rupture frequency and consequences has been substantially modified for the present version of NUREG–1150. Core damage frequencies obtained using the new analyses are consistent with the values cited above as results from other recent PRAs.

Comment: The models of core degradation are unrealistic. Severe fuel damage is defined to cover all cases where fuel cladding is damaged, but cladding integrity is not a measure of fuel damage and results in an overestimate of risk when it is defined as such.

Response:

The models that are used to predict fuel damage do not attempt to describe all the complex phenomena associated with severe core degradation in detail. The thermal-hydraulic model in the Source Term Code Package (STCP) (Ref. D.24) uses simplified models and assumptions for the treatment of some of the very complex steps in the core degradation process, such as fuel slumping into the lower plenum of a reactor vessel. However, the current version of NUREG–1150 did not rely heavily on the thermal-hydraulic model in the STCP for the estimation of the core meltdown process. The results of analysis with the MELCOR code (Ref. D.26), the MELPROG code (Ref. D.27), and the MAAP code (Ref. D.42) assisted project analysts and other experts in estimating the magnitude of parameters directly associated with core melt progression, such as hydrogen production and the mode of reactor vessel failure. Although the MELCOR code and the MELPROG code predict some core meltdown processes in more detail than the models in the STCP, the simplifications in the models of these codes must also be recognized.

In the current version of NUREG–1150, core damage is defined as a significant core uncovery occurrence with reflooding of the core not imminently expected. The result is a prolonged uncovery of the core that leads to damaged fuel and an expected release of fission products from the fuel.

The current version of NUREG–1150 treats the recovery of core with fuel damage differently from earlier probabilistic risk analyses. Under a broad range of conditions, given that a water supply is recovered prior to vessel failure, the likelihood of recovering a core and arresting an accident was evaluated. Based largely on the experience of the accident at Three Mile Island Unit 2, debris bed coolability analyses, and supplemental calculations of head failure, the likelihood of arresting further core damage decreased as the fraction of the fuel relocated to the bottom head at the time of coolant recovery increased.

D.6 Containment Loads and Structural Response

D.6.1 Accident Progression Event Tree: Logic

Comment: The large amount of detail in the accident progression event trees (APETs) gives the impression that more is known about containment events and phenomena than is actually known. It is difficult to believe the results of a complex tree that yields a tremendous number of pathways that are then aggregated into a dozen or so groups. Not only does a complex tree give a false impression but it limits any review of how the tree was constructed.

Response:

The rationale for developing a detailed APET is to provide an explicit treatment of all phenomena that can have a significant impact on the accident progression and the magnitude of the fission product source terms. Even if the existing capability to predict some of these phenomena is limited, it is important that the phenomena be recognized, at least for characterizing the uncertainty in the results.

The size of an APET does not affect the clarity of the results. Pathways with similar characteristics can be grouped to form simpler event trees as has been done to present results in the current version of the main report.