WASH – 1400
(NUREG-75/014)

Reactor Safety Study

An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants

Appendix V

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QUANTITATIVE RESULTS

of

ACCIDENT SEQUENCES

APPENDIX V

to

REACTOR SAFETY STUDY

U.S. NUCLEAR REGULATORY COMMISSION
OCTOBER 1975

Appendix V

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Section 1

Introduction

The purpose of this appendix is to present the methods used to quantify the accident sequences previously defined in Appendix I that can potentially lead to the release of significant amounts of radioactivity from nuclear power plants. The objective of this quantification was to provide the input information needed to perform calculations of the consequences of these accident sequences as described in Appendix VI.

Appendix I, which presents all the explicitly defined accident sequences developed for the PWR and BWR, actually started with the consideration of many thousands of potential sequences. As noted in Appendix I, sections 2.4 and 4.3, and the Addendum to the Main Report, the vast majority of these were eliminated as non-contributors because the accident sequences that remained are those indicated in the set of event trees for core accidents and the additional sequences for non-core accidents presented in Appendix I. This material systematically identified a collection of accident sequences potentially capable of resulting in significant releases of radioactive material from a nuclear power plant.

In principle, for each one of the accident sequences, one could compute its probability of occurrence, the amount of radioactive material released, as well as a few other factors that will be discussed later, and with this information one could compute consequences. One would then have for each accident sequence its probability and associated consequences. However, such exhaustive analysis is unnecessary because there are within this very large number of accident sequences, sets of sequences having similar consequences. Within each of these consequence sets, the probabilities of the individual sequences cover such a wide range that only a few of the sequences in each set determine the probability that the particular release will occur.

Hence, the accident sequences were grouped into sets determined by the magnitude of their associated radioactive release. These will be called release categories hereafter. This grouping was done by selecting, from each event tree constructed in Appendix I, those sequences that appeared to cover all important variations of ESF operability states and containment failure modes. These sequences will be called key sequences hereafter. The radioactive release magnitudes were then computed for each of the key sequences. Based on these computed releases, the magnitudes of each of the release categories was established and each key sequence was assigned to a release category. Each of the accident sequences from each event tree was then identified with a key sequence on the basis of engineering similitude and thus identified with one of the release categories.

Once the complete set of accident sequences was partitioned into release categories, it was necessary to determine the probability of occurrence of each release category. This was done by assigning roughly estimated probabilities to each sequence in each category. On this basis the few sequences that determined the probability of occurrence of each release category were selected. These will be called dominant sequences. The probabilities of the dominant sequences were then computed as precisely as could be done, including a careful reexamination of each of these sequences to ensure that potential common mode failure contributions were identified.[1]

The probability for each release category was then computed by combining the probabilities of the dominant accident sequences. Here again it is important to note that a complete probability assessment of each accident sequence was not required. Once the dominant sequences in each release category were identified, the other sequences were demonstrated not to contribute significantly to the overall probability by the use of bounding techniques as is discussed in section 4.1.1 of this appendix.[2]

The resulting set of release categories with their representative radioactive

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[1] See the Addendum to the Main Report and Appendix IV for discussion of potential common mode failures.

[2] The preceeding discussion is a simplified description of the actual procedure used. In practice, these steps involved many iterations in an attempt to ensure their completeness and accuracy.

release, probability values, and other necessary information (to be described in the next section) were used as input for the consequence model described in Appendix VI.

In addition to the accident sequences derived from the event trees of Appendix I, potential accidents from sources of radioactivity in locations other than the core were covered in an analysis described in section 5 of Appendix I.

Section 2

Summary of Results

The results of the quantification of the accident sequences of interest are presented in Table V 2-1, Summary of Accidents Involving Core, and in Table V 2-2, Summary of Accidents Not Involving Core. These tables also list additional specific input data needed for the calculation of consequences. The following information is presented in these tables:

a. Release Category - These are significantly different categories which were selected by the screening of key accident sequences to identify sequences that are significantly different from one another in terms of radioactivity released from the plant to the environment.

b. Probability per Year - This is the median of the probability of the occurrence of radioactive releases of the size identified with the particular release category and is obtained by combining (by Monte Carlo techniques) the probabilities of the highest probability sequences (dominant) in each release category. This median value also includes a 10% contribution from each adjacent category to account for uncertainties in the size of the release.

c. Time of Release - This is the time interval between the initiating event and the major release of radioactivity from the containment building. This is used in the consequence model to account for radioactive decay before the release occurs.

d. Warning Time for Evacuation - This is the time interval between the system(s) failure which will certainly result in a core melt accident and the release of a large amount of radioactive material from the containment. It is assumed that the signal to start evacuation will be given at the time it is certain that the core will melt.

e. Duration of Release - This is the period over which the bulk of the radioactive release takes place. It is used in the consequence model to account for wind meander.

f. Elevation of Release - This is the height above the ground of the point where the radioactivity is released from the containment building. It is one of the factors used in the consequence model to describe the dispersion of radioactivity.

g. Containment Energy Release - This is the estimated rate of sensible energy release that accompanies the major discharge of radioactivity from the containment. It is used in the consequence model to describe the supplementary vertical dispersion due to initial thermal energy in the release.

h. Fraction of Core Inventory Release - This is the fraction of the amounts of the most important element groups, present in the core at the time of the potential accident, that are released from the containment. It is used in the consequence model, in conjunction with core inventory data, to compute the actual amounts of radioactive isotopes released.

TABLE V 2-1 SUMMARY OF ACCIDENTS INVOLVING CORE

RELEASE CATEGORY	PROBABILITY per Reactor-Yr	TIME OF RELEASE (Hr)	DURATION OF RELEASE (Hr)	WARNING TIME FOR EVACUATION (Hr)	ELEVATION OF RELEASE (Meters)	CONTAINMENT ENERGY RELEASE (10^6 Btu/Hr)	FRACTION OF CORE INVENTORY RELEASED[a]							
							Xe-Kr	Org. I	I	Cs-Rb	Te-Sb	Ba-Sr	Ru[b]	La[c]
PWR 1	9×10^{-7}	2.5	0.5	1.0	25	520[d]	0.9	6×10^{-3}	0.7	0.4	0.4	0.05	0.4	3×10^{-3}
PWR 2	8×10^{-6}	2.5	0.5	1.0	0	170	0.9	7×10^{-3}	0.7	0.5	0.3	0.06	0.02	4×10^{-3}
PWR 3	4×10^{-6}	5.0	1.5	2.0	0	6	0.8	6×10^{-3}	0.2	0.2	0.3	0.02	0.03	3×10^{-3}
PWR 4	5×10^{-7}	2.0	3.0	2.0	0	1	0.6	2×10^{-3}	0.09	0.04	0.03	5×10^{-3}	3×10^{-3}	4×10^{-4}
PWR 5	7×10^{-7}	2.0	4.0	1.0	0	0.3	0.3	2×10^{-3}	0.03	9×10^{-3}	5×10^{-3}	1×10^{-3}	6×10^{-4}	7×10^{-5}
PWR 6	6×10^{-6}	12.0	10.0	1.0	0	N/A	0.3	2×10^{-3}	8×10^{-4}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	1×10^{-5}
PWR 7	4×10^{-5}	10.0	10.0	1.0	0	N/A	6×10^{-3}	2×10^{-5}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}
PWR 8	4×10^{-5}	0.5	0.5	N/A	0	N/A	2×10^{-3}	5×10^{-6}	1×10^{-4}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0	0
PWR 9	4×10^{-4}	0.5	0.5	N/A	0	N/A	3×10^{-6}	7×10^{-9}	1×10^{-7}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0	0
BWR 1	1×10^{-6}	2.0	2.0	1.5	25	130	1.0	7×10^{-3}	0.40	0.40	0.70	0.05	0.5	5×10^{-3}
BWR 2	6×10^{-6}	30.0	3.0	2.0	0	30	1.0	7×10^{-3}	0.90	0.50	0.30	0.10	0.03	4×10^{-3}
BWR 3	2×10^{-5}	30.0	3.0	2.0	25	20	1.0	7×10^{-3}	0.10	0.10	0.30	0.01	0.02	3×10^{-3}
BWR 4	2×10^{-6}	5.0	2.0	2.0	25	N/A	0.6	7×10^{-4}	8×10^{-4}	5×10^{-3}	4×10^{-3}	6×10^{-4}	6×10^{-4}	1×10^{-4}
BWR 5	1×10^{-4}	3.5	5.0	N/A	150	N/A	5×10^{-4}	2×10^{-9}	6×10^{-11}	4×10^{-9}	8×10^{-12}	8×10^{-14}	0	0

(a) A discussion of the isotopes used in the study is found in Appendix VI. Background on the isotope groups and release mechanisms is found in Appendix VII.

(b) Includes Mo, Rh, Tc, Co.

(c) Includes Nd, Y, Ce, Pr, La, Nb, Am, Cm, Pu, Np, Zr.

(d) A lower energy release rate than this value applies to part of the period over which the radioactivity is being released. The effect of lower energy release rates on consequences is found in Appendix VI.

TABLE V 2-2 SUMMARY OF ACCIDENTS NOT INVOLVING CORE[a]

ACCIDENT	PROBABILITY OF OCCURRENCE per Reactor-Yr	TIME OF RELEASE (Hr)	DURATION OF RELEASE (Hr)	ELEVATION OF RELEASE (M)	EQUIVALENT FRACTION OF CORE INVENTORY RELEASED[c]							
					Xe-Kr	Org. I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La
LOSS OF COOLING IN SFSP	$<10^{-6}$[b]	5	5	40	10^{-1}	7×10^{-4}	7×10^{-4}	10^{-3}	6×10^{-4}	10^{-4}	2×10^{-4}	2×10^{-5}
DROPPED SHIPPING CASK	6×10^{-7}	10	10	0	3×10^{-7}	10^{-11}	10^{-11}	1×10^{-7}	3×10^{-10}	5×10^{-11}	~0	~0
REFUELING ACCIDENT	10^{-3}	<1	<1	40	9×10^{-5}	4×10^{-7}	4×10^{-7}	6×10^{-9}	2×10^{-11}	3×10^{-11}	~0	~0
WASTE GAS STORAGE TANK RUPTURE	10^{-2}	<1	<1	0	2×10^{-4}	10^{-9}	10^{-9}	~0	~0	~0	~0	~0
LIQUID WASTE STORAGE TANK RUPTURE	10^{-2}	<1	<1	0	~0	8×10^{-8}	8×10^{-8}	6×10^{-8}	4×10^{-9}	5×10^{-11}	2×10^{-8}	10^{-11}
EARTHQUAKE-INDUCED LOSS OF COOLING IN SFSP WITH LOSS OF AIR COOLING SYSTEM	3×10^{-8}	5	5	0	10^{-1}	7×10^{-2}	7×10^{-2}	10^{-1}	6×10^{-2}	10^{-2}	10^{-2}	10^{-3}

(a) PWR and BWR designs were examined, and the more severe accidents were selected as representative bounds for both.
(b) Estimated probability includes consideration of turbine- and tornado-generated missiles.
(c) Fractions of total core inventory 30 minutes after shutdown.

Section 3

Discussion of Key Accident Sequences and Release Categories

The event trees and other techniques described in Appendix I systematically identified about 1000 explicitly defined accident sequences potentially capable of causing significant releases of radioactive material from PWR and BWR nuclear power plants.

Examination of both the ESF failures and the resulting physical processes involved in the various event tree sequences revealed patterns of similarity that could be used to characterize the spectrum of releases of radioactive materials from the plant. Recognition of these patterns suggested that the selection of sets of representative, or key, accident sequences to define the spectrum of releases was possible. Hence, it was possible to restrict the calculation of radioactive releases only to the key sequences. This representative group of sequences covered the significantly different 1) accident time histories, 2) system involvement in accident sequences, and 3) magnitudes of radioactive releases. Table 3-1 indicates the key accident sequences and the system and containment failure mode coverage obtained for the PWR and BWR.

Attachment 1 presents a detailed discussion of the quantification of the radioactive releases from each key accident sequence. The quantified releases were calculated by the CORRAL code and are presented in Attachment 1, Table 6, for the PWR key accident sequences and in Attachment 1, Table 13, for the BWR key accident sequences.[1] The CORRAL computer program models those containments which determine the behavior of radioactivity in the reactor containment. It is a multi-compartment model which represents the following processes: natural transport and deposition, removal of radioactivity by aqueous sprays, recirculation filter systems, once-through filter systems, water pool scrubbing, and leakage or exhaust from containment to the outside atmosphere. All these processes may occur simultaneously and the need for time dependent analyses required computer solution of the set of differential equations. [This computer program was given the code name CORRAL, and because of the containment differences between the PWR and the BWR, two versions were developed; that is, CORRAL-PWR and CORRAL-BWR.] The output from these programs is the calculated fraction of the core radioactive inventory which is released to the external atmosphere as a function of time for each key accident sequence.

The calculation of the radioactivity released from the containment barrier for any key accident sequence required input data obtained from various sources. These input data consist of the physical description of the containment, and of the physical phenomena occurring, as well as the amounts of radioactive materials released to the containment.

The physical description of the containment, that is, the number of compartments and their arrangement, volumes, surface areas, heights, and so forth, is determined by the type of reactor plant.

The physical phenomena that would be expected to occur during potential reactor meltdown accidents included the following:

a. The timetable of the accident, particularly the times for the initiation and completion of core melting.

b. Steam generation rates during core meltdown.

c. The rate and extent of zirconium-water reaction.

d. The probability and consequences of hydrogen burning or explosion in the containment building.

e. Probability and consequences of steam explosions in the reactor vessel when molten core materials come into contact with water; and the likelihood that such steam explosions would be energetic enough to rupture the reactor vessel in the containment building.

f. The time required for the molten core to melt through the reactor vessel.

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[1] CORRAL is described in Attachment 1 to this appendix. The "CORRAL User's Guide" is presented in Appendix VII.

g. Probability and consequences of steam explosions that could occur when the molten core falls to the floor of the reactor cavity; also the probability of containment failure due to the steam explosion.

h. Pressure-time histories within the reactor containment building, including the times of potential containment failure due to overpressure.

i. Time dependent leak rates from the containment building.

j. The interaction of the molten core with the concrete foundation and the time required to melt through the bottom of the containment building.

k. The anticipated movement of the molten core after the melt-through.

In order to estimate the radioactivity released from the core into the containment, applicable data on radioactive release, transport and deposition within the reactor systems were reviewed and applied in several areas of radioactivity behavior in containment systems. The method involved identifying four core material radioactive release terms for each of seven classes of radioactive species. The four basic release terms consisted of:

- The gap release component.
- The core melt release component.
- The vaporization release component.
- The core steam explosion release component.

The seven classes of radioactive species considered included the noble gases, the halogens (elemental and organic halides), the alkali metals, the tellurium group, the alkaline earths, the noble metals group and the refractory oxide group. The release values for each of these species are expressed as fractions of the initial core inventory. These values must be multiplied by the fraction of the core which participates in the release process or event of interest in order to develop the proper release terms.

After the input for each key accident sequence is complete and initial conditions have been set, the CORRAL code uses the data to continuously compute changing properties and fission product removal rates as a function of time. These values are used in incremental solutions to the coupled set of differential equations to obtain the time dependent concentrations and accumulations of radioactivity in each compartment of the containment.

The output from CORRAL calculations consists principally of cumulative fractional releases from containment vs. time for each of the various isotopic groups. The releases are identified as occurring at ground level or at an elevation, and the temperature of the released gases is provided to assist subsequent estimations of plume buoyancy. Actual accident time values are also specified in order that the radioactive decay corrections can be made.

The CORRAL calculated releases for the key accident sequences were used as preliminary input data to the consequence model. The consequence model predicted the results of key accident sequences in terms of fatalities, injuries, long term health effects, and property damage.

The results of the detailed CORRAL calculations and of the preliminary runs of the consequence model were examined for patterns involving the various ESF operability states and containment failure modes. Such patterns were found and these suggested that a grouping of key accident sequences into release categories established by these patterns could be made. The releases from several additional accident sequences were also calculated to ensure that all significant variations in ESF operability states and levels of release were included in the set of key accident sequences and release categories. Tables V 3-2 and V 3-3 show this grouping.

On the basis of the understanding gained of the impact of variations in ESF operability and containment failure modes on the magnitude of the radioactive release, it was then possible to allocate and group the balance of the accident sequences from the event trees into release categories. The allocation of the sequences was based on comparison of the phenomena involved in each sequence with the phenomena involved in the key sequences of the release categories. Tables V 3-4 to V 3-12 show this allocation for each of the PWR and BWR event trees.

Using probability data from the quantification of accident sequences (presented in section 4 of this appendix)

the dominant (i.e., most probable), accident sequences in each release category were readily identified.

The source terms for these dominant accident sequences were grouped together, and representative source terms for the release categories were conservatively selected. Table V 3-13 presents an example of the representative selection of a radioactive release for the PWR Release Category 2. The representative source terms for each of the release categories were previously shown in Table V 2-1 of this appendix.

The grouping of all accident sequences into release categories resulted in a restructuring of the problem into a form which showed clearly that, of all the postulated accidents leading to a given release, only a few dominant sequences made a significant contribution to the probability of occurrence of the release. Thus, the effort of estimation of releases and probabilities could concentrate on those sequences that were the main contributors to the risk. Tables V 3-14 to V 3-17 summarize the allocation of dominant accident sequences for each event tree to the release categories.

TABLE V 3-1 KEY ACCIDENT SEQUENCES FROM EVENT TREES

Sequence Designation		Event Tree Failure(s)	Containment Event Tree Failure
PWR Sequences			
1.	A	Large Rupture Only	None
2.	A-β	Large Rupture Only	Containment Leakage
3.	AH-α	ECCS Recirculation	Vessel Steam Explosion
4.	AH-β	ECCS Recirculation	Containment Leakage
5.	AH-ε	ECCS Recirculation	Melt-through
6.	AHI-α	ECCS Recirculation plus Sodium Hydroxide	Vessel Steam Explosion
7.	AHI-β	ECCS Recirculation plus Sodium Hydroxide	Containment Leakage
8.	AHI-ε	ECCS Recirculation plus Sodium Hydroxide	Melt-through
9.	AG-δ	Containment Heat Removal	Overpressure
10.	AHG-δ	ECCS Recirculation plus Containment Heat Removal	Overpressure
11.	AHG-ε	ECCS Recirculation plus Containment Heat Removal	Melt-through
12.	AHF-α	ECCS Recirculation plus Containment Spray Recirculation	Vessel Steam Explosion
13.	AHF-β	ECCS Recirculation plus Containment Spray Recirculation	Containment Leakage
14.	AHF-δ	ECCS Recirculation plus Containment Spray Recirculation	Overpressure
15.	AHF-ε	ECCS Recirculation plus Containment Spray Recirculation	Melt-through
16.	AD-α	ECCS Injection	Vessel Steam Explosion
17.	AD-β	ECCS Injection	Containment Leakage
18.	AD-ε	ECCS Injection	Melt-through
19.	ADI-α	ECCS Injection plus Sodium Hydroxide	Vessel Steam Explosion
20.	ADI-ε	ECCS Injection plus Sodium Hydroxide	Melt-through
21.	ADG-ε	ECCS Injection plus Containment Heat Removal	Melt-through

TABLE V 3-1 (continued)

Sequence Designation		Event Tree Failure(s)	Containment Event Tree Failure
22.	ADGI- ε	ECCS Injection plus Containment Heat Removal plus Sodium Hydroxide	Melt-through
23.	ADF- β	ECCS Injection plus Containment Spray Recirculation	Containment Leakage
24.	ADF- ε	ECCS Injection plus Containment Spray Recirculation	Melt-through
25.	ACD- β	Containment Spray Injection plus ECCS Injection	Containment Leakage
26.	ACD- ε	Containment Spray Injection plus ECCS Injection	Melt-through
27.	ACDGI- α	All Except Electric Power and Containment Spray Recirculation	Vessel Steam Explosion
28.	ACDGI- β	All Except Electric Power and Containment Spray Recirculation	Containment Leakage
29.	ACDGI- δ	All Except Electric Power and Containment Spray Recirculation	Overpressure
30.	ACDGI- ε	All Except Electric Power and Containment Spray Recirculation	Melt-through
31.	AB- α	Electric Power	Vessel Steam Explosion
32.	AB- γ	Electric Power	Hydrogen Combustion
33.	AB- ε	Electric Power	Melt-through
34.	S_2C-α	Small LOCA plus Containment Spray Injection	Vessel Steam Explosion
35.	S_2C-δ	Small LOCA plus Containment Spray Injection	Overpressure
36.	TMLB'-γ	Transient plus Feedwater plus Electric Power	Hydrogen Combustion
37.	TMLB'-δ	Transient plus Feedwater plus Electric Power	Overpressure
38.	TMLB'-α	Transient plus Feedwater plus Electric Power	Vessel Steam Explosion

TABLE V 3-1 (continued)

Sequence Designation	Event Tree Failure(s)	Containment Event Tree Failure
BWR Sequences		
1. AE-β (dry)	All Emergency Coolant Injection	Containment Steam Explosion
2. AE-γ (dry)	All Emergency Coolant Injection	Containment Overpressure[a]
3. AF-β	ECCS Functionability	Containment Steam Explosion
4. AF-γ	ECCS Functionability	Containment Overpressure
5. AF-α	ECCS Functionability	Vessel Steam Explosion
6. AJ-γ	Post Accident Heat Removal	Containment Overpressure
7. ADE-γ (dry)	All Emergency Coolant Injection plus Vapor Suppression	Containment Overpressure
8. ADF-γ	ECCS Functionability plus Vapor Suppression	Containment Overpressure
9. ADJ-γ	Post Accident Heat Removal plus Vapor Suppression	Containment Overpressure
10. AEG-δη (dry)	All Emergency Coolant Injection plus Containment Leakage	Drywell Leakage plus Sec. Containment Isolation
11. AEG-δ (dry)	All Emergency Coolant Injection plus Containment Leakage	Drywell Leakage Only
12. AGJ-δθ	Post Accident Heat Removal plus Containment Leakage	Drywell Leakage plus SGTS Filters
13. AGJ-δ	Post Accident Heat Removal plus Containment Leakage	Drywell Leakage Only
14. AE-α	Partial Emergency Coolant Injection	Vessel Steam Explosion
15. ADE-γ′ (dry)	All Emergency Coolant Injection plus Vapor Suppression	Containment Overpressure[b]
16. ADF-γ′	ECCS Functionability plus Vapor Suppression	Containment Overpressure
17. AJ-γ′	Post Accident Heat Removal	Containment Overpressure
18. ADJ-γ′	Post Accident Heat Removal plus Vapor Suppression	Containment Overpressure
19. A	Pipe Break Only	None
20. TC-α	Transient Requiring Shutdown plus Reactor Trip plus Liquid Poison Injection	Vessel Steam Explosion
21. TC-γ	Transient Requiring Shutdown plus Reactor Trip plus Liquid Poison Injection	Containment Overpressure

TABLE V 3-1 (continued)

Sequence Designation		Event Tree Failure(s)	Containment Event Tree Failure
22.	TW-α	Transient Shutdown plus Post Accident Heat Removal	Vessel Steam Explosion
23.	TW-γ	Transient Shutdown plus Post Accident Heat Removal	Containment Overpressure
24.	TW-γ'	Transient Shutdown plus Post Accident Heat Removal	Containment Overpressure

(a) γ is used to indicate that containment failure occurs by overpressurization with a release path where a large amount of deposition of radioactivity occurs.

(b) γ' is used to indicate that containment failure occurs by overpressure, but with a release path where no deposition of radioactivity occurs. Examination of the containment and reactor building layout indicated that the probability for this release path was about 0.2 for all overpressure failures.

TABLE V 3-2 PWR KEY ACCIDENT SEQUENCES ARRANGED IN RELEASE CATEGORIES

Release Categories								
← Major Consequences								Minor Consequences →
1	2	3	4	5	6	7	8	9
AHF-α	AB-γ	AD-α	ACD-β	AH-β	AHF-ε	AH-ε	A-β	A
ACDGI-α	AHF-δ	AH-α	ACDGI-β	AHI-β	ADF-ε	AHI-ε	AI-β	AI
AB-α	AHF-β	AHI-α	ADG-β	AD-β	AB-ε	AHG-ε		
AG-α	ADF-β	ADI-α				AHG-δ		
S_2C-α	TMLB'-γ	ADG-α				AD-ε		
TMLB'-α	TMLB'-δ	AG-δ				ADI-ε		
		S_2C-δ				ACD-ε		
						ACDGI-ε		
						ACDGI-δ		
						ADG-ε		

TABLE V 3-3 BWR KEY ACCIDENT SEQUENCES ARRANGED IN RELEASE CATEGORIES

Release Categories				
← major consequences				minor consequences →
1	2	3	4	5
AE-α	AE-β	AE-γ	AEJ-δ	A
TC-α	ADE-γ'	ADE-γ	AEG-δ	
TW-α	ADJ-γ'	ADJ-γ		
	AJ-γ'	AJ-γ		
	TW-γ'	AEG-δη		
		AEG-δθ		
		AGJ-δθ		
		TC-γ		
		TW-γ		

TABLE V 3-4 PWR LARGE LOCA ACCIDENT SEQUENCES vs. RELEASE CATEGORIES

Core melt | No core melt

Release Categories								
1	2	3	4	5	6	7	8	9
Dominant Large LOCA Accident Sequences With Point Estimates								
AB-α 1×10^{-11} AF-α 1×10^{-10} ACD-α 5×10^{-11} AG-α 9×10^{-11}	AB-γ 1×10^{-10} AHF-γ 2×10^{-11} AB-δ 4×10^{-11}	AD-α 2×10^{-8} AH-α 1×10^{-8} AF-δ 1×10^{-8} AG-δ 9×10^{-9}	ACD-β 1×10^{-11}	AD-β 4×10^{-9} AH-β 3×10^{-9}	AB-ε 1×10^{9} ADF-ε 2×10^{-10} AHF-ε 1×10^{-10}	AD-ε 2×10^{-6} AH-ε 1×10^{-6}	A-β 2×10^{-7}	A 1×10^{-4}
Other Large LOCA Accident Sequences								
ACDGI-α AHFI-α ACHF-α ACDI-α ACDG-α AGI-α AFI-α ACG-α ACGI-α ACF-α ACDF-α ACEI-α ACEG-α ACEGI-α ACEF-α ACE-α AHF-α	ADF-β AHFI-δ ACHF-δ ACHF-γ ACDF-γ ACEF-γ AHFI-β ADFI-β ACHF-β ACDF-β AHF-δ AHFI-γ AEF-β AEFI-β ACEF-β AEF-δ AEFI-δ ACEF-δ AB-β AHF-β	AHG-α AHGI-α ADF-α ADFI-α ACH-α ACHI-α ACHG-α ACHGI-α AGI-δ AFI-δ ACG-δ ACGI-δ ACF-δ AHI-α ADGI-α ADI-α ADG-α AE-α AEI-α AEF-α AEFI-α AEG-α AEGI-α	ACDGI-β ADG-β ACDI-β ACDG-β ADGI-β ACE-β ACEI-β ACEG-β ACEGI-β AEG-β AEGI-β	AHI-β AHG-β AHGI-β ADI-β ACH-β ACHI-β ACHG-β AE-β AEI-β	ACHGI-ε AHFI-ε ADFI-ε ACDF-ε ACDGI-ε ACHF-ε AEF-ε AEFI-ε ACEF-ε ACEGI-ε	AHG-δ AHGI-δ AHGI-ε ACH-ε ACHI-ε ACHG-δ ACHG-ε ACHGI-ε ACDI-ε ACDG-δ ACDG-ε ADG-δ ADGI-δ AHG-ε ADI-ε ADG-ε ACD-ε ADGI-ε AHI-ε AE-ε AEI-ε ACE-ε ACEI-ε ACEG-ε ACEG-δ ACEGI-δ ACHGI-δ AEG-δ AEGI-δ AEG-ε AEGI-ε	AI-β AC-β ACI-β	AI AC ACI
$A\Sigma_P$ [a]	3×10^{-10}	2×10^{-10}	5×10^{-8}	1×10^{-11}	7×10^{-9}	1×10^{-9}	3×10^{-6}	2×10^{-7}	1×10^{-4}

[a] Σ_p is the arithmetic sum of the probabilities of the accident sequence in each release category.

TABLE V 3-5 PWR SMALL LOCA S_1 ACCIDENT SEQUENCES vs. RELEASE CATEGORIES

Core Melt | No Core Melt

Release Categories								
1	2	3	4	5	6	7	8	9
Dominant Small LOCA S_1 Accident Sequences								
$S_1B-\alpha$ $3x10^{-11}$ $S_1CD-\alpha$ $7x10^{-11}$ $S_1F-\alpha$ $3x10^{-10}$ $S_1G-\alpha$ $3x10^{-10}$	$S_1B-\gamma$ $4x10^{-10}$ $S_1HF-\gamma$ $6x10^{-11}$ $S_1B-\delta$ $1x10^{-10}$	$S_1D-\alpha$ $3x10^{-8}$ $S_1H-\alpha$ $3x10^{-8}$ $S_1F-\delta$ $3x10^{-8}$ $S_1G-\delta$ $3x10^{-8}$	$S_1CD-\beta$ $1x10^{-11}$	$S_1H-\beta$ $5x10^{-9}$ $S_1D-\beta$ $6x10^{-9}$	$S_1DF-\varepsilon$ $3x10^{-10}$ $S_1B-\varepsilon$ $2x10^{-9}$ $S_1HF-\varepsilon$ $4x10^{-10}$	$S_1D-\varepsilon$ $3x10^{-6}$ $S_1H-\varepsilon$ $3x10^{-6}$	$S_1-\beta$ $6x10^{-7}$	S_1 $3x10^{-4}$
Other Small LOCA S_1 Accident Sequences								
$S_1HFI-\alpha$ $S_1CHF-\alpha$ $S_1CDI-\alpha$ $S_1CDGI-\alpha$ $S_1CDG-\alpha$ $S_1CDF-\alpha$ $S_1KC-\alpha$ $S_1KCG-\alpha$ $S_1KCF-\alpha$ $S_1BK-\alpha$ $S_1FI-\alpha$ $S_1GI-\alpha$ $S_1CG-\alpha$ $S_1CGI-\alpha$ $S_1CF-\alpha$ $S_1HF-\alpha$	$S_1HFI-\delta$ $S_1HFI-\beta$ $S_1CHF-\beta$ $S_1CHF-\gamma$ $S_1CHF-\delta$ $S_1CDF-\gamma$ $S_1CDF-\beta$ $S_1KF-\beta$ $S_1KFI-\beta$ $S_1KCF-\gamma$ $S_1KCF-\beta$ $S_1BK-\gamma$ $S_1BK-\beta$ $S_1DFI-\beta$ $S_1B-\beta$ $S_1HF-\beta$ $S_1DF-\beta$ $S_1HF-\delta$	$S_1HI-\alpha$ $S_1CDGI-\delta$ $S_1HG-\alpha$ $S_1GI-\delta$ $S_1HGI-\alpha$ $S_1FI-\delta$ $S_1DI-\alpha$ $S_1DG-\alpha$ $S_1DG-\delta$ $S_1DF-\alpha$ $S_1DFI-\alpha$ $S_1CH-\alpha$ $S_1CHI-\alpha$ $S_1CG-\delta$ $S_1CHG-\alpha$ $S_1CGI-\delta$ $S_1CHGI-\alpha$ $S_1CF-\gamma$ $S_1K-\alpha$ $S_1KI-\alpha$ $S_1KG-\alpha$ $S_1KGI-\alpha$ $S_1KF-\alpha$ $S_1KFI-\alpha$	$S_1DG-\beta$ $S_1DGI-\beta$ $S_1CDI-\beta$ $S_1CDGI-\beta$ $S_1CDG-\beta$ $S_1KG-\beta$ $S_1KGI-\beta$ $S_1KC-\beta$ $S_1KCG-\beta$	$S_1HI-\beta$ $S_1HG-\beta$ $S_1HGI-\beta$ $S_1DI-\beta$ $S_1CH-\beta$ $S_1CHI-\beta$ $S_1CHG-\beta$ $S_1CHGI-\beta$ $S_1KI-\beta$ $S_1K-\beta$	$S_1HFI-\varepsilon$ $S_1DFI-\varepsilon$ $S_1CHF-\varepsilon$ $S_1KCF-\varepsilon$ $S_1BK-\varepsilon$	$S_1CH-\varepsilon$ $S_1CD-\varepsilon$ $S_1HI-\varepsilon$ $S_1HG-\delta$ $S_1HG-\varepsilon$ $S_1HGI-\varepsilon$ $S_1DI-\varepsilon$ $S_1DG-\varepsilon$ $S_1DGI-\varepsilon$ $S_1CHI-\varepsilon$ $S_1CHG-\varepsilon$ $S_1CHG-\delta$ $S_1CHGI-\delta$ $S_1CHGI-\varepsilon$ $S_1CDI-\varepsilon$ $S_1CDGI-\varepsilon$ $S_1CDG-\varepsilon$ $S_1K-\varepsilon$ $S_1KI-\varepsilon$ $S_1KG-\varepsilon$ $S_1KGI-\varepsilon$ $S_1KC-\varepsilon$ $S_1KCG-\varepsilon$ $S_1KCG-\delta$	$S_1I-\beta$ $S_1C-\beta$ $S_1CI-\beta$	S_1I S_1C S_1CI
$S_1\Sigma_p$ 7×10^{-10}	7×10^{-10}	1×10^{-7}	1×10^{-11}	1×10^{-9}	3×10^{-9}	6×10^{-6}	6×10^{-7}	3×10^{-4}

TABLE V 3-6 PWR SMALL LOCA S_2 ACCIDENT SEQUENCES vs. RELEASE CATEGORIES

	Core Melt							No Core Melt	
	Release Categories								
	1	2	3	4	5	6	7	8[a]	9[a]
	Dominant Small LOCA S_2 Accident Sequences								
	$S_2B-\alpha$ $1x10^{-10}$ $S_2F-\alpha$ $1x10^{-9}$ $S_2CD-\alpha$ $1x10^{-10}$ $S_2G-\alpha$ $9x10^{-10}$ $S_2C\alpha$ $2x10^{-8}$	$S_2B-\gamma$ $1x10^{-9}$ $S_2HF-\gamma$ $2x10^{-10}$ $S_2B-\delta$ $4x10^{-10}$	$S_2D-\alpha$ $9x10^{-8}$ $S_2H-\alpha$ $6x10^{-8}$ $S_2F-\delta$ $1x10^{-7}$ $S_2C-\delta$ $2x10^{-6}$ $S_2G-\delta$ $9x10^{-8}$	$S_2DG-\beta$ $1x10^{-12}$	$S_2D-\beta$ $2x10^{-8}$ $S_2H-\beta$ $1x10^{-8}$	$S_2B-\varepsilon$ $8x10^{-9}$ $S_2CD-\varepsilon$ $2x10^{-8}$ $S_2HF-\varepsilon$ $1x10^{-9}$	$S_2D-\varepsilon$ $9x10^{-6}$ $S_2H-\varepsilon$ $6x10^{-6}$		
	Other Small LOCA S_2 Accident Sequences								
	$S_2HFI-\alpha$ $S_2LC-\alpha$ $S_2KC-\alpha$ $S_2BK-\alpha$ $S_2GI-\alpha$ $S_2FI-\alpha$ $S_2HF-\alpha$	$S_2HFI-\delta$ $S_2HFI-\beta$ $S_2KF-\beta$ $S_2KFI-\beta$ $S_2KC-\gamma$ $S_2KC-\beta$ $S_2BK-\gamma$ $S_2BK-\beta$ $S_2DFI-\beta$ $S_2LF-\beta$ $S_2LFI-\beta$ $S_2LC-\beta$ $S_2CD-\beta$ $S_2CD-\delta$ $S_2LC-\delta$ $S_2HF-\beta$ $S_2DF-\beta$ $S_2B-\beta$ $S_2HF-\delta$	$S_2HI-\alpha$ $S_2GI-\delta$ $S_2HG-\alpha$ $S_2GI-\delta$ $S_2HGI-\delta$ $S_2FI-\delta$ $S_2DI-\alpha$ $S_2DG-\alpha$ $S_2DG-\delta$ $S_2DF-\alpha$ $S_2DFI-\alpha$ $S_2CD-\alpha$ $S_2K-\alpha$ $S_2KI-\alpha$ $S_2KG-\alpha$ $S_2KGI-\alpha$ $S_2KF-\alpha$ $S_2KFI-\alpha$ $S_2L-\alpha$ $S_2LI-\alpha$ $S_2LG-\alpha$ $S_2LGI-\alpha$ $S_2LF-\alpha$ $S_2LFI-\alpha$	$S_2DGI-\beta$ $S_2LG-\beta$ $S_2LGI-\beta$ $S_2KG-\beta$ $S_2KGI-\beta$	$S_2HI-\beta$ $S_2HG-\beta$ $S_2HGI-\beta$ $S_2DI-\beta$ $S_2L-\beta$ $S_2LI-\beta$ $S_2K-\beta$ $S_2KI-\beta$	$S_2HFI-\varepsilon$ $S_2DF-\varepsilon$ $S_2DFI-\varepsilon$ $S_2LF-\varepsilon$ $S_2LFI-\varepsilon$ $S_2BK-\varepsilon$ $S_2KF-\varepsilon$ $S_2KFI-\varepsilon$ $S_2KC-\varepsilon$	$S_2HI-\varepsilon$ $S_2HG-\varepsilon$ $S_2HG-\delta$ $S_2HGI-\varepsilon$ $S_2DI-\varepsilon$ $S_2DG-\varepsilon$ $S_2DGI-\varepsilon$ $S_2L-\varepsilon$ $S_2LI-\varepsilon$ $S_2LG-\varepsilon$ $S_2LGI-\varepsilon$ $S_2K-\varepsilon$ $S_2KI-\varepsilon$ $S_2KG-\varepsilon$ $S_2KGI-\varepsilon$		
$S_2\Sigma_p$	3×10^{-8}	2×10^{-9}	3×10^{-6}	1×10^{-12}	3×10^{-8}	3×10^{-8}	2×10^{-5}		

(a) No sequences in these categories are shown since negligible radioactivity release is expected to occur when all ESFs properly operate.

TABLE V 3-7 PWR TRANSIENT SEQUENCES vs. RELEASE CATEGORIES

						Core Melt	No Core Melt	
Release Categories								
1	2	3	4	5	6	7	8	9
Dominant PWR Transient Accident Sequences								
TMLB'-α[a] $3x10^{-8}$	TMLB'-γ $7x10^{-7}$ TMLB'-δ $2x10^{-6}$	TML-α $6x10^{-8}$ TKQ-α $3x10^{-8}$ TKMQ-α $1x10^{-8}$		TML-β $3x10^{-10}$ TKQ-β $3x10^{-10}$	TMLB'-ε $6x10^{-7}$	TML-ε $6x10^{-6}$ TKQ-ε $3x10^{-6}$ TKMQ-ε $1x10^{-6}$		
Other Transient Accident Sequences								
TMLC'-α TMLF'-α TMLG'-α TKQF-α TKQB-α TKQG-α TKMQF-α TKMQB-α TKMQG-α TMLPC'-α TMLPB'-α TMLPF'-α TMLPG'-α TKPC-α TKPF-α TKPB-α TKPG-α TKQUC-α TKQUF-α TKQUG-α TKMUC-α TKMUF-α TKMUB-α TKMUG-α TKMPC-α TKMPF-α TKMPB-α TKMPG-α TKMQUC-α TKMQUF-α TKMQUG-α TKMQUB-α TKMLC-α TKMLF-α TKMLB-α TKMLG-α TMLQUC'-α TMLQUB'-α TMLQUF'-α TMLQUG'-α TKMLPC-α TKMLPB-α	TMLC'-γ TMLC'-δ TMLF'-γ TMLF'-δ TMLG'-γ TMLG'-δ TKMQF-γ TKMQF-δ TKMQB-γ TKMQB-δ TKMQG-γ TKMQG-δ TKQC-γ TKQC-δ TKQG-γ TKQG-δ TKQF-γ TKQF-δ TKQB-γ TKQB-δ TLMPC'-γ TLMPC'-δ TLMPB'-γ TLMPB'-δ TLMPG'-δ TLMPG'-γ TLMPF'-γ TLMPF'-δ TKPC-γ TKPF-γ TKPB-γ TKPG-γ TKPC-δ TKPF-δ TKPB-δ TKPG-δ TKQUC-δ TKQUF-δ TKQUB-δ TKQUG-δ TKQUC-γ TKQUF-γ	TMLP-α TKP-α TKQU-α TKMU-α TKMP-α TKMQU-α TKML-α TMLQU-α TKMLP-α TMLQ-α		TMLP-β TKP-β TKQU-β TKMU-β TKMQU-β TKML-β TMLQU-β TKMLP-β TKMQC-β TMLQ-β	TMLC'-ε TMLF'-ε TMLG'-ε TKQF-ε TKQG-ε TKQB-ε TKMQF-ε TKMQG-ε TKMQB-ε TMLPB'-ε TMLPC'-ε TMLPF'-ε TMLPG'-ε TKPB-ε TKPC-ε TKPF-ε TKPG-ε TKQUB-ε TKQUC-ε TKQUF-ε TKQUG-ε TKMUC-ε TKMUB-ε TKMUG-ε TKMUF-ε TKMPC-ε TKMPB-ε TKMPG-ε TKMPF-ε TKMQUC-ε TKMQUB-ε TKMQUF-ε TKMQUG-ε TKMLC-ε TKMLB-ε TKMLF-ε TKMLG-ε TMLQUC'-ε TMLQUB'-ε TMLQUF'-ε TMLQUG'-ε TKMLPC-ε	TMLP-ε TKP-ε TKQU-ε TKMU-ε TKMQU-ε TKMP-ε TKML-ε TMLQU-ε TKMLP-ε TMLQ-ε		

TABLE V 3-7 (continued)

	Core Melt ⎮ No Core Melt								
	Release Categories								
	1	2	3	4	5	6	7	8	9
	Other Transient Accident Sequences								
	TKMLPF-α TKMLPG-α TKQC-α TKMQC-α TMLQB-α TMLQC-α TMLQF-α TMLQG-α	TKQUB-γ TKQUG-γ TKMUC-γ TKMUF-γ TKMUB-γ TKMUG-γ TKMUC-δ TKMUF-δ TKMUG-δ TKMUB-δ TKMPC-δ TKMPF-δ TKMPB-δ TKMPG-δ TKMPC-γ TKMPF-γ TKMPB-γ TKMPG-γ TKMQUC-γ TKMQUF-γ TKMQUB-γ TKMQUG-γ TKMQUC-δ TKMQUF-δ TKMQUB-δ TKMQUG-δ TKMLC-δ TKMLB-δ TKMLF-δ TKMLG-δ TKMLC-γ TKMLB-γ TKMLF-γ TKMLG-γ TMLQUC'-γ TMLQUB'-γ TMLQUF'-γ TMLQUG'-γ TMLQUC'-δ TMLQUB'-δ TMLQUF'-δ TMLQUG'-δ TKMLPC-δ TKMLPB-δ TKMLPF-δ TKMLPG-δ TKMLPC-γ TKMLPB-γ TKMLPF-γ TKMLPG-γ TKMQC-γ TKMQC-δ	TMLQC-γ TMLQF-γ TMLQB-γ TMLQG-γ TMLQC-δ TMLQF-δ TMLQG-δ TMLQB-δ TMLQC'-γ TMLQC'-δ TMLQB'-γ TMLQB'-δ TMLQG'-δ TMLQG'-γ TMLQF'-γ TMLQF'-δ			TKMLPF-ε TKMLPB-ε TKMLPG-ε TKQC-ε TKMQC-ε TMLQB-ε TMLQC-ε TMLQG-ε TMLQF-ε			
TI_p	3×10^{-8}	3×10^{-6}	1×10^{-7}		6×10^{-10}	6×10^{-7}	1×10^{-5}		

(a)Characters with primes (') indicate that the initiating event may influence system operability.

TABLE V 3-8

PART A - PWR ACCIDENT SEQUENCES FOR REACTOR VESSEL RUPTURE vs. RELEASE CATEGORIES

	Release Categories								
	1	2	3	4	5	6	7	8	9
	Reactor Vessel Rupture Sequences								
	RC-α $2x10^{-12}$ RB-α $2x10^{-14}$	RC-γ $3x10^{-11}$ RF-δ $1x10^{-11}$ RC-δ $1x10^{-12}$ RC-β 5×10^{-13}	R-α $1x10^{-9}$				R-ε $1x10^{-7}$		
$R\Sigma_p$	$2x10^{-12}$	$4x10^{-11}$	$1x10^{-9}$				$1x10^{-7}$		

PART B - PWR ACCIDENT SEQUENCES FOR RUPTURES INTO INTERFACING SYSTEMS vs. RELEASE CATEGORIES

	Release Categories								
	1	2	3	4	5	6	7	8	9
	Interfacing Systems LOCA Sequences (Check Valve)								
		V $4x10^{-6}$							
$V\Sigma_p$		$4x10^{-6}$							

TABLE V 3-9 BWR LARGE LOCA ACCIDENT SEQUENCES vs. RELEASE CATEGORIES

	Core melt				No Core melt
	Release Categories				
	1	2	3	4	5
	Dominant Large LOCA Accident Sequences				
	AE-α 2×10^{-9} AJ-α 1×10^{-10} AHI-α 1×10^{-10} AI-α 1×10^{-10}	AE-γ' 3×10^{-8} AE-β 1×10^{-8} AJ-γ'' 2×10^{-9} AI-γ'' 2×10^{-9} AHI-γ'' 2×10^{-9}	AE-γ 1×10^{-7} AJ-γ 1×10^{-8} AI-γ 1×10^{-8} AHI-γ 1×10^{-8}	AGJ-δ 6×10^{-11} AEG-δ 7×10^{-10} AGHI-δ 6×10^{-11}	A 1×10^{-4}
	Other Large LOCA Accident Sequences				
	AFG-α ADF-α AC-α ACG-α ACD-α AHJ-α AGJ-α AGI-α AGHJ-α AGHI-α AEG-α wet ADJ-α ADI-α ADHJ-α ADHI-α ADE-α wet AF-α	AE-β wet AB-β ABG-β AHI-β AGJ β AGHJ-β AGHI-β AEG-β wet AEG-β dry ACG-β AFG-β AF-β AF-γ'' AC-β ADE-γ'' ADJ-γ'' AC-γ'' ADF-γ'' ACD-γ'' AGI-β	AG-γ'' ADE-γ ADF-γ ADJ-γ AE-γ wet AB-γ ACD-γ AHJ-γ ADE-γ wet ABD-γ ADHI-γ ADI-γ ADHJ-γ AF-γ AEG-δη AEG-δζ wet AEG-δζ dry ABG-δζ AGHI-δζ AEG-εζ wet AEG-εζ dry ABG-εζ AGHI-εζ AGJ-δζ AGI-δζ AGHJ-δζ AGJ-δη AGI-δη AGHJ-δη AGJ-εζ AGI-εη AGHJ-εη AEG-δη wet AEG-δθ wet AEG-δθ dry AEG-εη wet AEG-εη dry AEG-εθ wet AEG-εθ dry ABC-δη ABG-εη AEG-εη AGHI-δη AGHI-δθ AGHI-εη AGHI-εθ AGI-δθ AGHJ-δθ AGJ-εθ AGI-εθ AGHJ-εθ	AFG-δζ AFG-δη AFG-δθ AFG-δ ACG-δζ ACG-δη ACG-δθ ACG-δ AFG-εζ AFG-εη AFG-εθ AFG-ε ACG-εζ ACG-εη ACG-εθ ACG-ε AGI-δ AGHJ-ε AGJ-ε AGI-ε AGHJ-ε AEG-δ wet AEG-ε dry AEG-ε wet AGHI-ε	
$A\Sigma_p$	2×10^{-9}	4×10^{-8}	2×10^{-7}	8×10^{-10}	1×10^{-4}

Table V 3-8 — Table V 3-9

TABLE V 3-10 BWR SMALL LOCA ACCIDENT SEQUENCES (S_1) VS. RELEASE CATEGORIES

Core Melt | No Core Melt

Release Categories				
1	2	3	4	5
Dominant Small LOCA Accident Sequences (S_1)				
S_1E-α 2×10^{-9} S_1J-α 3×10^{-10} S_1I-α 4×10^{-10} S_1HI-α 4×10^{-10}	S_1E-γ' 4×10^{-8} S_1E-β 1×10^{-8} S_1J-γ' 7×10^{-9} S_1I-γ' 7×10^{-9} S_1HI-γ' 6×10^{-9}	S_1E-γ 1×10^{-7} S_1J-γ 3×10^{-8} S_1HI-γ 2×10^{-8} S_1I-γ 4×10^{-8} S_1C-γ 3×10^{-9}	S_1GJ-δ 2×10^{-10} S_1GI-δ 2×10^{-10} S_1EG-δ 1×10^{-10} S_1GHI-δ 2×10^{-10}	
Other Small LOCA Accident Sequences (S_1)				
S_1C-α S_1CG-α S_1CD-α S_1HJ-α S_1GJ-α S_1GI-α S_1GHJ-α S_1GHI-α S_1EG-α (wet) S_1DJ-α S_1DI-α S_1DHJ-α S_1DHI-α S_1DE-α (wet) S_1B-α S_1BG-α S_1BD-α	S_1BG-β (wet) S_1B-β (wet) S_1GHI-β S_1GJ-β S_1GI-β S_1GHI-β S_1CG-β S_1C-β S_1BG-β S_1EG-β S_1DE-γ' S_1DJ-γ' S_1C-γ' S_1CD-γ' S_1HJ-γ' S_1DI-γ' S_1DHJ-γ' S_1DHI-γ' S_1B-γ' S_1BD-γ' S_1HI-β	S_1HJ-γ S_1DJ-γ S_1DI-γ S_1DHJ-γ S_1DHI-γ S_1DE-γ (wet) S_1CD-γ S_1B-γ S_1BD-γ S_1EG-δζ (wet) S_1BG-δζ S_1GHI-δζ S_1EG-εζ (wet) S_1BG-εζ S_1GHI-εζ S_1GJ-δζ S_1GI-δζ S_1GHJ-δζ S_1GJ-δη S_1GI-δη S_1GHJ-δη S_1GJ-εζ S_1GI-εζ S_1GHJ-εζ S_1GJ-εη S_1GI-εη S_1CHJ-εη S_1EG-δη (wet) S_1EG-δθ (wet) S_1EG-εη (wet) S_1EG-εθ (wet) S_1BG-δη S_1GHI-δη S_1GHI-δθ S_1GHI-εη S_1GHI-εθ S_1BG-εη S_1GI-δθ S_1GHI-δθ S_1GJ-εθ S_1GI-εθ S_1GHJ-εθ	S_1CG-δζ S_1CG-δη S_1CG-δθ S_1CG-δ S_1CG-εζ S_1CG-εη S_1CG-εθ S_1CG-ε S_1GHJ-δ S_1GHJ-ε S_1GJ-ε S_1GI-ε S_1EG-ε S_1GHI-ε	
$S_1\Sigma_p$ 3×10^{-9}	7×10^{-8}	2×10^{-7}	7×10^{-10}	

TABLE V 3-11 BWR SMALL LOCA ACCIDENT SEQUENCES (S_2) VS. RELEASE CATEGORIES

	Core Melt				No Core Melt
	Release Categories				
	1	2	3	4	5
	Dominant Small LOCA Accident Sequences (S_2)				
	$S_2J-\alpha$ 1×10^{-9} $S_2I-\alpha$ 1×10^{-9} $S_2HI-\alpha$ 1×10^{-9} $S_2E-\alpha$ 5×10^{-10}	$S_2E-\gamma'$ 1×10^{-8} $S_2E-\beta$ 4×10^{-9} $S_2J-\gamma'$ 2×10^{-8} $S_2I-\gamma'$ 2×10^{-8} $S_2HI-\gamma'$ 2×10^{-8}	$S_2E-\gamma$ 4×10^{-8} $S_2J-\gamma$ 8×10^{-8} $S_2I-\gamma$ 9×10^{-8} $S_2HI-\gamma$ 9×10^{-8} $S_2C-\gamma$ 8×10^{-9}	$S_2CG-\delta$ 6×10^{-11} $S_2GHI-\delta$ 6×10^{-10} $S_2EG-\delta$ 3×10^{-10} $S_2GJ-\delta$ 6×10^{-10} $S_2GI-\delta$ 2×10^{-10}	
	Other Small LOCA Accident Sequences (S_2)				
	$S_2HJ-\alpha$ $S_2GJ-\alpha$ $S_2GI-\alpha$ $S_2GHJ-\alpha$ $S_2GHI-\alpha$ $S_2EC-\alpha$ (wet) $S_2DJ-\alpha$ $S_2DI-\alpha$ $S_2DHJ-\alpha$ $S_2DHI-\alpha$ $S_2DE-\alpha$ (wet) $S_2B-\alpha$ $S_2BG-\alpha$ $S_2BD-\alpha$ $S_2CD-\alpha$ $S_2CG-\alpha$ $S_2C-\alpha$	$S_2BG-\beta$ (wet) $S_2B-\beta$ (wet) $S_2GJ-\beta$ $S_2GI-\beta$ $S_2GHI-\beta$ $S_2HI-\beta$ $S_2GHJ-\beta$ $S_2CG-\beta$ $S_2C-\beta$ $S_2EG-\beta$ $S_2DE-\gamma'$ $S_2DJ-\gamma'$ $S_2C-\gamma'$ $S_2CD-\gamma'$ $S_2HJ-\gamma'$ $S_2DI-\gamma'$ $S_2DHJ-\gamma'$ $S_2B-\gamma'$ $S_2BD-\gamma'$ $S_2DHI-\gamma'$	$S_2HJ-\gamma$ $S_2DJ-\gamma$ $S_2DI-\gamma$ $S_2DHJ-\gamma$ $S_2DHI-\gamma$ $S_2DE-\gamma$ (wet) $S_2CD-\gamma$ $S_2B-\gamma$ $S_2BD-\gamma$ $S_2GJ-\delta\zeta$ $S_2GI-\delta\zeta$ $S_2GHJ-\delta\zeta$ $S_2GJ-\delta\eta$ $S_2GI-\delta\eta$ $S_2GHJ-\delta\eta$ $S_2GJ-\varepsilon\zeta$ $S_2GI-\varepsilon\zeta$ $S_2GHJ-\varepsilon\zeta$ $S_2GI-\varepsilon\theta$ $S_2GHJ-\varepsilon\theta$ $S_2GJ-\varepsilon\eta$ $S_2GI-\varepsilon\eta$ $S_2GHJ-\varepsilon\eta$ $S_2EG-\delta\eta$ (wet) $S_2EG-\delta\theta$ (wet) $S_2EG-\varepsilon\eta$ (wet) $S_2EG-\varepsilon\theta$ (wet) $S_2BG-\delta\eta$ $S_2CHI-\delta\eta$ $S_2GHI-\delta\theta$ $S_2GHI-\varepsilon\eta$ $S_2GHI-\varepsilon\theta$ $S_2BG-\varepsilon\eta$ $S_2GI-\delta\theta$ $S_2GHI-\delta\theta$ $S_2GJ-\varepsilon\theta$	$S_2CG-\delta\zeta$ $S_2CG-\delta\eta$ $S_2CG-\delta\theta$ $S_2CG-\varepsilon\zeta$ $S_2CG-\varepsilon\eta$ $S_2CG-\varepsilon\theta$ $S_2CG-\varepsilon$ $S_2EG-\varepsilon$ $S_2GHI-\varepsilon$ $S_2GI-\varepsilon$ $S_2GJ-\varepsilon$ $S_2GHJ-\varepsilon$ $S_2GHJ-\delta$	
$S_2\Sigma_p$	4×10^{-9}	7×10^{-8}	4×10^{-7}	2×10^{-9}	

TABLE V 3-12 BWR TRANSIENT ACCIDENT SEQUENCES vs. RELEASE CATEGORIES

	Release Categories			Core Melt	No Core Melt
	1	2	3	4	5
	Dominant Transient Accidents				
	TW-α $2x10^{-7}$ TC $1x10^{-2}$ $1x10^{-7}$ TQUV-α $5x10^{-9}$	TW-γ' $3x10^{-6}$ TQUV-γ' $8x10^{-8}$	TW-γ $1x10^{-5}$ TC-γ $1x10^{-5}$ TQUV-γ $4x10^{-7}$		
	Other Transient Accidents				
	TQW-α TQUW-α TPW-α TPQW-α TPQUW-α	TQW-γ' TQUW-γ' TPW-γ' TPQW-γ' TPQUW-γ' TPQUV-γ'	TQW-γ TQUW-γ TPW-γ TPQW-γ TPQUW-γ TPQUV-γ		
TI_p	$3x10^{-7}$	$3x10^{-6}$	$2x10^{-5}$		

TABLE V 3-13 ILLUSTRATIVE EXAMPLE SHOWING COMPARISON OF RELEASES FROM KEY PWR ACCIDENT SEQUENCES AND SELECTION OF REPRESENTATIVE VALUES FOR PWR RELEASE CATEGORY 2

Time (Hours)	Cumulative Fractions of Core Inventory Released to the Atmosphere								Event
	Xe-Kr	Org-I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La	
					Case ABγ				
0.27	$2x10^{-6}$	0	$3x10^{-6}$	$9x10^{-6}$	$7x10^{-8}$	$1x10^{-9}$	0	0	Just Before Melt
1.0	$1x10^{-4}$	$9x10^{-7}$	$1x10^{-4}$	$1x10^{-4}$	$2x10^{-5}$	$1x10^{-5}$	$3x10^{-6}$	$3x10^{-7}$	End Melt Release
1.0	$8x10^{-1}$	$6x10^{-3}$	$6x10^{-1}$	$4x10^{-1}$	$8x10^{-2}$	$6x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	Overpressure
2.5	$9x10^{-1}$	$7x10^{-3}$	$7x10^{-1}$	$5x10^{-1}$	$1x10^{-1}$	$6x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	1/2 Vapor Release
8.3	$9x10^{-1}$	$7x10^{-3}$	$7x10^{-1}$	$5x10^{-1}$	$1x10^{-1}$	$6x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	
720	$9x10^{-1}$	$7x10^{-3}$	$7x10^{-1}$	$5x10^{-1}$	$1x10^{-1}$	$6x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	
					Case AHF-δ				
1.6	$2x10^{-5}$	0	$2x10^{-6}$	$2x10^{-6}$	$2x10^{-8}$	$3x10^{-10}$	0	0	Just Before Melt
3.8	$7x10^{-4}$	$5x10^{-6}$	$4x10^{-4}$	$2x10^{-4}$	$7x10^{-5}$	$3x10^{-5}$	$1x10^{-5}$	$1x10^{-6}$	Just Before OP
3.8	$8x10^{-1}$	$6x10^{-3}$	$3x10^{-1}$	$2x10^{-1}$	$3x10^{-1}$	$2x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	Overpressure
5.5	$8x10^{-1}$	$6x10^{-3}$	$3x10^{-1}$	$2x10^{-1}$	$3x10^{-1}$	$2x10^{-2}$	$2x10^{-2}$	$4x10^{-3}$	End Vapor Release
720	1.0	$7x10^{-3}$	$4x10^{-1}$	$2x10^{-1}$	$3x10^{-1}$	$2x10^{-2}$	$2x10^{-2}$	$4x10^{-3}$	
					Case AHFβ				
1.6	$1x10^{-2}$	0	$1x10^{-3}$	$9x10^{-5}$	$7x10^{-7}$	$1x10^{-8}$	0	0	Just Before Melt
2.5	$2x10^{-1}$	$1x10^{-3}$	$1x10^{-1}$	$9x10^{-2}$	$2x10^{-2}$	$1x10^{-2}$	$4x10^{-3}$	$4x10^{-4}$	End Melt Release
5.5	$8x10^{-1}$	$6x10^{-3}$	$4x10^{-1}$	$3x10^{-1}$	$3x10^{-1}$	$4x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	End Vapor Release
720	1.0	$7x10^{-3}$	$4x10^{-1}$	$3x10^{-1}$	$3x10^{-1}$	$4x10^{-2}$	$3x10^{-2}$	$4x10^{-3}$	
					Case ADFβ				
0.25	$2x10^{-3}$	0	$8x10^{-4}$	$8x10^{-3}$	$1x10^{-5}$	$2x10^{-7}$	0	0	Just Before Melt
1.0	$7x10^{-2}$	$3x10^{-4}$	$5x10^{-3}$	$1x10^{-2}$	$3x10^{-3}$	$2x10^{-3}$	$5x10^{-4}$	$5x10^{-5}$	End Melt Release
4.0	$6x10^{-1}$	$2x10^{-3}$	$3x10^{-2}$	$6x10^{-2}$	$2x10^{-2}$	$5x10^{-3}$	$1x10^{-2}$	$2x10^{-3}$	End Vapor Release
26.5	$8x10^{-1}$	$3x10^{-3}$	$4x10^{-2}$	$7x10^{-2}$	$2x10^{-1}$	$6x10^{-3}$	$1x10^{-2}$	$3x10^{-3}$	
720	1.0	$4x10^{-3}$	$5x10^{-2}$	$7x10^{-2}$	$2x10^{-1}$	$6x10^{-3}$	$1x10^{-2}$	$3x10^{-3}$	

TABLE V 3-13 (Continued)

Time (Hours)	Cumulative Fractions of Core Inventory Released to the Atmosphere								Event
	Xe-Kr	Org-I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La	
				Case TLMB'γ					
2.6	$7x10^{-4}$	0	$4x10^{-4}$	$8x10^{-4}$	$2x10^{-6}$	$2x10^{-8}$	0	0	Just Before Melt
3.6	$2x10^{-2}$	$1x10^{-4}$	$1x10^{-2}$	$1x10^{-2}$	$2x10^{-3}$	$1x10^{-3}$	$3x10^{-4}$	$3x10^{-5}$	End Melt Rel.
3.7	$8x10^{-1}$	$6x10^{-3}$	$6x10^{-1}$	$4x10^{-1}$	$8x10^{-2}$	$5x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	Overpressure
6.7	$8x10^{-1}$	$6x10^{-3}$	$7x10^{-1}$	$4x10^{-1}$	$1x10^{-1}$	$6x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	End Vap. Rel.
17	$9x10^{-1}$	$7x10^{-3}$	$7x10^{-1}$	$5x10^{-1}$	$2x10^{-1}$	$6x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	
720	1.0	$7x10^{-3}$	$7x10^{-1}$	$5x10^{-1}$	$2x10^{-1}$	$6x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	
				Case TMLB'δ					
2.6	$7x10^{-4}$	0	$4x10^{-4}$	$8x10^{-4}$	$2x10^{-6}$	$2x10^{-8}$	0	0	Just Before Melt
3.6	$2x10^{-2}$	$1x10^{-4}$	$1x10^{-2}$	$1x10^{-2}$	$2x10^{-3}$	$1x10^{-3}$	$3x10^{-4}$	$3x10^{-5}$	End Melt Rel.
3.7	$8x10^{-1}$	$6x10^{-3}$	$6x10^{-1}$	$4x10^{-1}$	$8x10^{-2}$	$5x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	Overpressure
6.7	$8x10^{-1}$	$6x10^{-3}$	$6x10^{-1}$	$4x10^{-1}$	$1x10^{-1}$	$5x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	End Vap. Rel.
17	$9x10^{-1}$	$7x10^{-3}$	$6x10^{-1}$	$4x10^{-1}$	$2x10^{-1}$	$5x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	
720	1.0	$7x10^{-3}$	$7x10^{-1}$	$4x10^{-1}$	$2x10^{-1}$	$5x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	
				Representative Release: Cat. #2-PWR[a]					
2.5	.9	$7x10^{-3}$	.7	.5	.3	.06	.02	$4x10^{-3}$	

(a) The underlined values are the representative releases selected from key sequences. Generally these values were the largest releases in each category of isotopes during the early time period when the radioactivity that had the maximum public impact was released to the atmosphere.

TABLE V 3-14 PWR DOMINANT ACCIDENT SEQUENCES vs. RELEASE CATEGORIES

	RELEASE CATEGORIES						Core Melt	No Core Melt	
	1	2	3	4	5	6	7	8	9
LARGE LOCA A	AB-α 1×10^{-11} AF-α 1×10^{-10} ACD-α 5×10^{-11} AC-α 9×10^{-11}	AB-γ 1×10^{-10} AB-δ 4×10^{-11} AMF-γ 2×10^{-11}	AD-α 3×10^{-8} AH-α 1×10^{-8} AF-δ 1×10^{-8} AG-δ 9×10^{-9}	ACD-δ 1×10^{-11}	AD-δ 4×10^{-9} AH-δ 3×10^{-9}	AB-ε 1×10^{-9} AMF-ε 1×10^{-10} ADF-ε 2×10^{-10}	AD-ε 2×10^{-6} AH-ε 1×10^{-6}	A-β 2×10^{-7}	A 1×10^{-4}
A Probabilities	2×10^{-9}	1×10^{-8}	1×10^{-7}	1×10^{-8}	4×10^{-8}	3×10^{-7}	3×10^{-6}	1×10^{-5}	1×10^{-4}
SMALL LOCA S_1	S_1B-α 3×10^{-11} S_1CD-α 7×10^{-11} S_1F-α 1×10^{-10} S_1G-α 3×10^{-10}	S_1B-γ 4×10^{-10} S_1B-δ 1×10^{-10} S_1HF-γ 6×10^{-11}	S_1D-α 3×10^{-8} S_1H-α 3×10^{-8} S_1F-δ 3×10^{-8} S_1G-δ 3×10^{-8}	S_1CD-δ 1×10^{-11}	S_1H-δ 5×10^{-9} S_1D-δ 6×10^{-9}	S_1DF-ε 3×10^{-10} S_1B-ε 2×10^{-9} S_1HF-ε 4×10^{-10}	S_1D-ε 3×10^{-6} S_1H-ε 3×10^{-6}	S_1-β 6×10^{-7}	S_1 3×10^{-6}
S_1 Probabilities	3×10^{-9}	2×10^{-8}	2×10^{-7}	3×10^{-8}	8×10^{-8}	6×10^{-7}	6×10^{-6}	3×10^{-5}	3×10^{-4}
SMALL LOCA S_2	S_2B-α 1×10^{-10} S_2F-α 1×10^{-9} S_2CD-α 2×10^{-10} S_2G-α 9×10^{-10} S_2C-α 2×10^{-8}	S_2B-γ 1×10^{-9} S_2HF-γ 2×10^{-10} S_2B-δ 4×10^{-10}	S_2D-α 9×10^{-8} S_2H-α 6×10^{-8} S_2F-δ 1×10^{-7} S_2C-δ 2×10^{-6} S_2G-δ 9×10^{-8}	S_2OG-δ 1×10^{-12}	S_2D-δ 2×10^{-8} S_2H-δ 1×10^{-8}	S_2B-ε 8×10^{-9} S_2CD-ε 2×10^{-8} S_2HF-ε 1×10^{-9}	S_2D-ε 9×10^{-6} S_2H-ε 6×10^{-6}		
S_2 Probabilities	1×10^{-7}	3×10^{-7}	3×10^{-6}	3×10^{-7}	3×10^{-7}	2×10^{-6}	2×10^{-5}		
REACTOR VESSEL RUPTURE - R	RC-α 2×10^{-12}	RC-γ 3×10^{-11} RF-δ 1×10^{-11} RC-δ 1×10^{-12}	R-α 1×10^{-9}				R-ε 1×10^{-7}		
R Probabilities	2×10^{-11}	1×10^{-10}	1×10^{-9}	2×10^{-10}	1×10^{-9}	1×10^{-8}	1×10^{-7}		
INTERFACING SYSTEMS LOCA (CHECK VALVE) - V		V 4×10^{-6}							
V Probabilities	4×10^{-7}	4×10^{-6}	4×10^{-7}	4×10^{-8}					
TRANSIENT EVENT - T	TWLB'-α 3×10^{-8}	TWLB'-γ 7×10^{-7} TWLB'-δ 2×10^{-6}	TWL-α 6×10^{-8} TWQ-α 3×10^{-8} TWWQ-α 1×10^{-8}		TWL-δ 3×10^{-10} TWQ-δ 3×10^{-10}	TWLB'-ε 6×10^{-7}	TWL-ε 6×10^{-6} TWQ-ε 3×10^{-6} TWWQ-ε 1×10^{-6}		
T Probabilities	3×10^{-7}	3×10^{-6}	4×10^{-7}	7×10^{-8}	2×10^{-7}	2×10^{-6}	1×10^{-5}		
(Σ) SUMMATION OF ALL ACCIDENT SEQUENCES PER RELEASE CATEGORY									
MEDIAN (50% VALUE)	9×10^{-7}	8×10^{-6}	4×10^{-6}	5×10^{-7}	7×10^{-7}	6×10^{-6}	4×10^{-5}	4×10^{-5}	4×10^{-4}
LOWER BOUND (5% VALUE)	9×10^{-8}	8×10^{-7}	6×10^{-7}	9×10^{-8}	2×10^{-7}	2×10^{-6}	1×10^{-5}	4×10^{-6}	4×10^{-5}
UPPER BOUND (95% VALUE)	9×10^{-6}	8×10^{-5}	4×10^{-5}	5×10^{-6}	4×10^{-6}	3×10^{-5}	3×10^{-4}	4×10^{-4}	4×10^{-3}

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

TABLE V 3-15 KEY TO PWR ACCIDENT SEQUENCE SYMBOLS

A - Intermediate to large LOCA.

B - Failure of electric power to ESFs.

B' - Failure to recover either onsite or offsite electric power within about 1 to 3 hours following an initiating transient which is a loss of offsite AC power.

C - Failure of the containment spray injection system.

D - Failure of the emergency core cooling injection system.

F - Failure of the containment spray recirculation system.

G - Failure of the containment heat removal system.

H - Failure of the emergency core cooling recirculation system.

K - Failure of the reactor protection system.

L - Failure of the secondary system steam relief valves and the auxiliary feedwater system.

M - Failure of the secondary system steam relief valves and the power conversion system.

Q - Failure of the primary system safety relief valves to reclose after opening.

R - Massive rupture of the reactor vessel.

S_1 - A small LOCA with an equivalent diameter of about 2 to 6 inches.

S_2 - A small LOCA with an equivalent diameter of about 1/2 to 2 inches.

T - Transient event.

V - LPIS check valve failure.

α - Containment rupture due to a reactor vessel steam explosion.

β - Containment failure resulting from inadequate isolation of containment openings and penetrations.

γ - Containment failure due to hydrogen burning.

δ - Containment failure due to overpressure.

ε - Containment vessel melt-through.

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

TABLE V 3-16 BWR DOMINANT ACCIDENT SEQUENCES OF EACH EVENT TREE vs. RELEASE CATEGORY

	Core Melt				No Core Melt
	RELEASE CATEGORIES				
	1	2	3	4	5
LARGE LOCA DOMINANT ACCIDENT SEQUENCES (A)	AE-α 2×10^{-9} AJ-α 1×10^{-10} AHI-α 1×10^{-10} AI-α 1×10^{-10}	AE-γ′ 3×10^{-8} AE-δ 1×10^{-8} AJ-γ′ 2×10^{-9} AI-γ′ 2×10^{-9} AHI-γ′ 2×10^{-9}	AE-γ 1×10^{-7} AJ-γ 1×10^{-8} AI-γ 1×10^{-8} AHI-γ 1×10^{-8}	AGJ-δ 6×10^{-11} AEG-δ 7×10^{-10} AGHI-δ 6×10^{-11}	A 1×10^{-4}
A Probabilities	8×10^{-9}	6×10^{-8}	2×10^{-7}	2×10^{-8}	1×10^{-4}
SMALL LOCA DOMINANT ACCIDENT SEQUENCES (S_1)	S_1E-α 2×10^{-9} S_1J-α 3×10^{-10} S_1I-α 4×10^{-10} S_1HI-α 4×10^{-10}	S_1E-γ′ 4×10^{-8} S_1E-δ 1×10^{-8} S_1J-γ′ 7×10^{-9} S_1I-γ′ 7×10^{-9} S_1HI-γ′ 6×10^{-9}	SE-γ 1×10^{-7} S_1J-γ 3×10^{-8} S_1I-γ 4×10^{-8} S_1HI-γ 2×10^{-8} S_1C-γ 3×10^{-9}	S_1GJ-δ 2×10^{-10} S_1CI-δ 2×10^{-10} S_1EI-δ 1×10^{-10} S_1GHI-δ 2×10^{-10}	
S_1 Probabilities	1×10^{-8}	9×10^{-8}	2×10^{-7}	2×10^{-8}	
SMALL LOCA DOMINANT ACCIDENT SEQUENCES (S_2)	S_2J-α 1×10^{-9} S_2I-α 1×10^{-9} S_2HI-α 1×10^{-9} S_2E-α 5×10^{-10}	S_2E-γ′ 1×10^{-8} S_2E-δ 4×10^{-9} S_2J-γ′ 2×10^{-8} S_2I-γ′ 2×10^{-8} S_2HI-γ′ 2×10^{-8}	S_2E-γ 4×10^{-8} S_2J-γ 8×10^{-8} S_2I-γ 9×10^{-8} S_2HI-γ 9×10^{-8} S_2C-γ 8×10^{-9}	S_2CG-δ 6×10^{-11} S_2GHI-δ 6×10^{-10} S_2EG-δ 3×10^{-10} S_2GJ-δ 6×10^{-10} S_2CI-δ 2×10^{-10}	
S_2 Probabilities	2×10^{-8}	1×10^{-7}	4×10^{-7}	4×10^{-8}	
TRANSIENT DOMINANT ACCIDENT SEQUENCES (T)	TW-α 2×10^{-7} TC-α 1×10^{-7} TQUV-α 5×10^{-9}	TW-γ′ 3×10^{-6} TQUV-γ′ 8×10^{-8}	TW-γ 1×10^{-5} TC-γ 1×10^{-5} TQUV-γ 4×10^{-7}		
T Probabilities	1×10^{-6}	6×10^{-6}	2×10^{-5}	2×10^{-6}	
PRESSURE VESSEL RUPTURE ACCIDENTS (R)		P.V. RUPT. 1×10^{-8} Oxidizing Atmosphere	P.V. RUPT. 1×10^{-7} Non-oxidizing Atmosphere		
R Probabilities	2×10^{-9}	2×10^{-8}	1×10^{-7}	1×10^{-8}	
SUMMATION OF ALL ACCIDENT SEQUENCES PER RELEASE CATEGORIES					
MEDIAN (50% VALUE)	1×10^{-6}	6×10^{-6}	2×10^{-5}	2×10^{-6}	1×10^{-4}
LOWER BOUND (5% VALUE)	1×10^{-7}	1×10^{-6}	3×10^{-6}	5×10^{-7}	1×10^{-5}
UPPER BOUND (95% VALUE)	8×10^{-6}	3×10^{-5}	8×10^{-5}	1×10^{-5}	1×10^{-3}

Table V 3-16

TABLE V 3-17 KEY TO BWR ACCIDENT SEQUENCE SYMBOLS

A - Rupture of reactor coolant boundary with an equivalent diameter of greater than six inches.

B - Failure of electric power to ESFs.

C - Failure of the reactor protection system.

D - Failure of vapor suppression.

E - Failure of emergency core cooling injection.

F - Failure of emergency core cooling functionability.

G - Failure of containment isolation to limit leakage to less than 100 volume per cent per day.

H - Failure of core spray recirculation system.

I - Failure of low pressure recirculation system.

J - Failure of high pressure service water system.

M - Failure of safety/relief valves to open.

P - Failure of safety/relief valves to reclose after opening.

Q - Failure of normal feedwater system to provide core make-up water.

S_1 - Small pipe break with an equivalent diameter of about 2"-6".

S_2 - Small pipe break with an equivalent diameter of about 1/2"-2".

T - Transient event.

U - Failure of HPCI or RCIC to provide core make-up water.

V - Failure of low pressure ECCS to provide core make-up water.

W - Failure to remove residual core heat.

α - Containment failure due to steam explosion in vessel.

β - Containment failure due to steam explosion in containment.

γ - Containment failure due to overpressure - release through reactor building.

γ' - Containment failure due to overpressure - release direct to atmosphere.

δ - Containment isolation failure in drywell.

ε - Containment isolation failure in wetwell.

ζ - Containment leakage greater than 2400 volume per cent per day.

η - Reactor building isolation failure.

θ - Standby gas treatment system failure.

Section 4

Quantification of Probabilities for Release Categories

4.1 BASIC CONSIDERATIONS

Given the accident sequences which were identified by the event trees and other models and given the individual system analysis obtained from the fault trees and system models, the accident sequences were then quantified to obtain the accident sequence probabilities. For a given accident sequence, the appropriate probability for the initiating event, the probability for the system (or systems) failing, and the probability for the containment failure mode as presented in attachment 1 to this appendix were all combined in order to obtain the accident sequence probability.

A particular accident sequence can be broken into three general factors - the occurrence of the initiating event, the failure of particular safeguard system or systems, and the containment failure mode:

$$\text{Accident Sequence} = \text{Initiating Event} \times \text{System(s) Failure} \times \text{Containment Failure Mode}$$

For the system failures, it is understood that certain systems also have succeeded.

The probability of the accident sequence is the probability of the combination of the three factors on the right hand side of the above expression. To obtain the correct accident sequence probability, the correct probabilities for the individual factors must be used, incorporating any dependencies among the factors. If independent probabilities are naively used, incorrect results can be obtained.

Using the above accident sequence expression, the accident sequence probability can be written as

$$P = P_{IE} \times P_{SF_1} \times \ldots \times P_{SF_n} \times P_{CFM},$$

where P denotes the probability and the subscripts are defined as follows: IE, initiating event; SF, system failure given the initiating event; CFM, containment failure mode given the initiating event and system failure.

The above equation accounts for the dependencies since each probability which is multiplied on the right hand side is not an independent probability but a dependent or conditional probability.

The second factor, the system(s) probability, is not independent but is the probability under the condition that the initiating event has occurred. As indicated earlier in Appendices I and II, the accident sequences prescribed by the event trees defined the conditions under which the fault trees for individual system fault trees were constructed. It should be noted that the conditioning limited the scope of the fault tree investigations to just those failure modes of interest for the event tree. This reduced the amount of potential dependent interreaction explorations by a large factor.

The probabilities of system failures, given the initiating event, need now to be discussed. As described in Appendix IX, and elsewhere, engineered safety features are required to be designed to operate despite following the occurrence of the initiating events, for example pipe event has occurred and the relevant systems have failed.[1] (There is an additional factor implied, which is the probability of the other systems in the sequence succeeding; however this can be ignored since these success probabilities are approximately 1.)

To quantify the accident sequence values for the above three probabilities - initiating event, system, and containment mode - under the defined conditions, the individual probabilities must be obtained and multiplied together. In the

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[1] In terms of general Boolean notation, the accident sequence can be expressed as ACCIDENT SEQUENCE = A • B • C where A,B,C denote the three factors (initiating event, etc.). The correct probability is thus P(accident sequence) = P(A)P(B/A)P(C/BA), which corresponds to the above equation. (P(B/A) is read as the probability of system failure given the initiating event.)

study, the initiating event probability was obtained directly from the data assessment values in Appendix III. The system failure probabilities, given the initiating event, were obtained from the fault trees and system models. Further discussion of this factor will be presented in the next paragraph. The containment failure mode probability, given the initiating event and relevant system failures, was obtained by analysis in Appendix VIII of the physical processes involved in the various accident sequences. The containment event trees incorporated the given conditions, or dependencies, in the model construction and definition, as discussed in Appendix I, and hence the correct dependent (conditional) probabilities were obtained directly.[1]

The probabilities of system failures, given the initiating event, need now to be discussed. As described in Appendix IX, and elsewhere, engineered safety features are required to be designed to operate despite the occurrence of the initiating events, for example pipe rupture. Thus, the system failure rate would be expected to be independent of the initiating event. However, to ensure incorporating potential dependencies, all fault trees and system models were constructed under the condition that the appropriate initiating event had occurred. When the initiating event caused certain components to be failed or to be ineffective, then these were so treated in the fault tree and system models. If the initiating event could cause or could accelerate failures, then they were explicitly incorporated in the models and subsequent quantifications. In addition, any dependencies or common modes, due to the initiating event, were overlayed on the analyses and quantifications.[2] Thus, the correct individual system probabilities were obtained which could then be used in the accident sequence probability equation given above.

If the event sequence contained a single system failure, the fault tree or system failure probability number could be used directly, and multiplication by the initiating event probability and containment mode failure probability then gave the particular accident sequence probability. In the use of this failure probability, a fault tree check was required to determine if the probability number needed to be modified to account for the other systems in the sequence which were defined as operating. Since the models and quantifications already incorporated the dominant effects of these constraints, no modification in general was required and the individual numbers were used directly. (For example, the fault trees incorporated the condition that electric power was not completely failed.)[1]

In the event a sequence contained two or more system failures, the individual system fault trees needed to be combined and the resulting Boolean expression reduced to extract the component failures which were common to the systems. Since two or more system failures can be treated as simply another fault tree with the individual systems coupled by an AND gate, the same techniques as used in the quantification of an individual system fault tree were used for the quantification of combined system fault trees. These techniques and quantification methods are more fully described in Appendix II.

Two examples of PWR accident sequences are described below to further illustrate the various dependencies that are included in the sequence quantifications.

<u>Example 1: ACD-α (PWR Release Category 1)</u>

This potential accident sequence is initiated by a large LOCA (A). During the course of the sequence both the emergency core cooling injection system (D) and the containment spray injection system (C) can fail. As a result of these potential failures, the core can melt and potentially cause a steam explosion (α) in the reactor vessel. This explosion may have sufficient energy to fail the vessel and have the upper part of the vessel penetrate the containment structure.

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[1] These values are presented in Tables 2 and 9 of Attachment 1 to this appendix.

[2] See example of accident sequence quantifications at the end of this section.

[1] For modification of independent-type fault trees, components which are common to systems operating and systems failing to operate would require appropriate conditional probabilities. Components (or critical paths) that are defined as operating by the accident sequence would therefore be deleted from the fault tree for system failure.

As noted in section 5.3.5 of Appendix IV, a potential common mode failure was identified that could contribute to the combined A and D events given that the large LOCA occurred at a specific location in the main loop piping. This common mode involved the possibility that, if a large rupture were to occur in the discharge of reactor coolant system pump B, it could cause the pump to overspeed, potentially resulting in a pump flywheel missile that might rupture the single injection line for the low-pressure injection system, thus fail D. The common mode failure contribution for this specific break location was assessed to be $\sim 1.3 \times 10^{-6}$ per reactor-year as the median probability for the AD events. On the other hand, the events A and D can be considered to be independent for all other break locations; this would result in a median probability value of about 6×10^{-7} per reactor-year.

Thus, the overall median probability of the AD event becomes about $(1.3 \times 10^{-6} + 5.6 \times 10^{-7}) \approx 2 \times 10^{-6}$ per reactor-year. This demonstrates that if independence between the A and D events had been assumed without investigating possible common mode dependencies between the initiating event and the systems, an underestimate would have resulted for all the accident sequences involving the AD combinations.

Analyses indicated that those rupture sizes characterized by the small LOCAs (i.e. < 6 inches in equivalent diameter) would not be expected to cause the RCS pump to overspeed, and because of this the same common mode potential did not apply to the small LOCAs. Also, the investigation of possible common mode dependencies between the C and D events (given the occurrence of either a large or a small LOCA) did not reveal any significant contributors.[1] The failure of the containment spray injection system (C) was thus taken to be independent of the AD events, and the median probability of its failure was taken to be about 2.4×10^{-3} per LOCA demand. This value for C was derived by use of fault tree analyses (presented in Appendix II), which also included the consideration of common mode failure possibilities identified at the system and component levels.

The median probability of a vessel steam explosion (α) given the large LOCA (A), the core-cooling injection system failure (D), and the ensuing core melt processes was assessed to be about 10^{-2} per core melt event.[1]

By simple combination of the above values, the overall median probability of 5×10^{-11} per reactor-year for the accident sequence ACD-α can be seen:

$$AD \times C \times \alpha = ACD\text{-}\alpha$$

$$(2 \times 10^{-6}) \times (2.4 \times 10^{-3}) \times (10^{-2}) \approx 5 \times 10^{-11}$$

Example 2: S_2 C-δ (PWR Release Category 3)

This accident sequence is one of interest because it was found to make a significant contribution to the probability of a core melt and to potentially large releases of radioactive materials. This potential sequence is initiatind by a small LOCA (S_2). During the course of the sequence, failure of the containment spray injection system (CSIS) can occur. Because of an identified dependency, failure of CSIS could result in the failure of both the containment spray recirculation system (CSRS) and the containment heat removal system (CHRS). This could cause the containment to fail in an overpressure mode (δ) after several hours. During this interval the emergency core cooling injection (ECI) and recirculation (ECR) systems are assumed to have been successfully operating to cool the core. On containment failure (at a pressure of approximately 100 psia[2]), the coolant residing in the containment, which is at an elevated temperature, would flash and cause the ECR pumps to cavitate and fail. The core could then melt in a containment that has already failed.

Investigation of potential system-to-system dependencies arising from various small-LOCA locations revealed that if the small LOCA (S_2) occurred, for example, in the region of the reactor vessel cavity, there would not be adequate communication for the spilled coolant to

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[1] See section 5.3.2 of Appendix IV.

[1] See Appendix VIII and attachment 1 to this appendix for discussions and results of the probability of vessel steam explosions.

[2] See Appendix VIII and Attachment 1 to this appendix for results of containment failure due to the S_2C-δ accident sequence.

flow between the cavity region and the containment "sump" used by the containment spray recirculation system (CSRS).

Since the CSRS pumps would be automatically actuated after the LOCA occurrence, they could fail shortly after they were started if the supply of water in the sump were inadequate. Thus, it appeared that successful operation of CSRS could occur only if the containment spray injection system (CSIS) were to operate and thus provide a supply of water to the sump.

These dependencies were illustrated in the small-LOCA (S_2) event tree.[1] The median probability value for the small LOCA (S_2) was assessed at approximately 10^{-3} per reactor-year since a large amount of small piping is located in the reactor vessel cavity region. The median unavailability of the containment spray injection system was derived by fault tree analyses to be about 2.4×10^{-3}, and the probability for containment failure via the overpressure mode (δ) was estimated to be near unity since CSRS failure would also cause loss of containment heat removal capability. These values combined to yield a median probability value of about (10^{-3}) (2.4×10^{-3}) $(\sim 1.0) = 2 \times 10^{-6}$ per reactor-year for the S_2C-δ sequence.

4.1.1 THE ROLE OF RELEASE CATEGORIES IN THE QUANTIFICATION

The grouping of the accident sequences into release categories played an important role in simplifying the quantification of the probability of occurrence of each category. The probabilities which were required for the final consequence calculations were the total probabilities associated with each of the various release categories which had been defined. The accident sequences had been grouped into discrete release categories as described earlier; hence the probability for a particular release category consisted simply of the sum of the individual accident sequence probabilities for those accident sequences lying in the same category.

[1]See CSRS and CSIS descriptions in section 4.1 of Appendix I and fault tree analyses in Appendix II.

Release Category	P(Release Category)
Accident Seq. 1	= P(Accident Seq. 1)
Accident Seq. 2	+ P(Accident Seq. 2)
.	.
.	.
.	.
Accident Seq. N	+ P(Accident Seq. N)

To facilitate the calculation of the release category probabilities, the dominant accident sequence or sequences were first selected for each category. This was done by examination of the individual system failure probabilities and included selection of the single system failure sequences. Because of the sequence breakdowns, the dominant sequences were quite apparent. These dominant sequences, once identified, were then quantified using the techniques described above, and these probabilities formed the first estimates for the release category probabilities.

Having calculated the dominant accident sequence probabilities, the remaining sequences in the category were then checked to determine if they could impact on the probability already computed for the release category. This checking involved the assignment of a maximum possible value (upper bound) for the sequence probability and determining whether any change in the release category probability resulted. In the assignment of the maximum values, the bounding techniques described in Appendix IV, section 3, were applied (i.e., when coupling could exist between systems, for example due to common components, the single system failure probability was used to calculate the maximum sequence probability, thereby assuming one to one coupling).[1]

In checking such possible impacts, if any effect was determined, the sequence probability was then precisely computed and added to the category probability. However, because a few single system failure sequences in all the categories dominated the probability by order of magnitude, the remaining sequences were determined to have negligible contributions. The checking thus saved the Boolean reduction which would have been

[1]In applying the bounding techniques of Appendix IV, a component failure as termed there is taken here as an individual system failure, and a combination of component failures is a combination of system failures in the sequence.

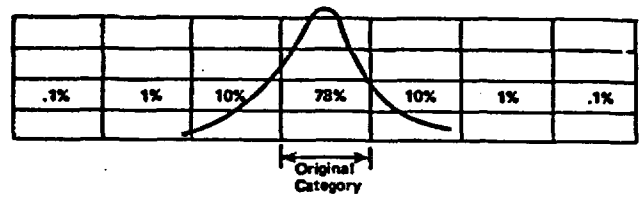
required on many of the multiple system failures. In addition, because of the checking and the dominance of single system failures, any effect of common modes between systems was verified to be negligible (at the maximum, common modes can change multiple system failures into single system failures; however, these already existed and dominated).

4.1.2 INCORPORATION OF RELEASE VARIATION (Sequence Smoothing)

The radioactive release magnitudes that were computed for each key accident sequence used best estimate values for the parameters describing the physical processes that could occur. These parameters included the radioactive release from the fuel, the temperatures and pressure accompanying the accident, washout efficiencies, and a number of other characteristics. In reality, due to the variability inherent in them, these physical processes would have not a single value but rather a distribution of possible values. Because of this variability, a given accident sequence will in reality have a spectrum of releases and hence a distribution of release fractions for each isotope.

This distribution of release fractions that can occur has a direct implication on the release categorization that is used in the study. Accident sequences were assigned to a release category based on the best estimate of release magnitudes.

To account for the distribution possible in release magnitudes, the study assigned a 10% chance that the sequence would lie in an adjacent release category, a 1% chance that it would lie in the next adjacent category, etc. These decreasing probability values were selected by the physical analysts performing the calculations. Only order-of-magnitude accuracies were needed for these values because of the larger error spreads on the calculated accident sequence probabilities. Accounting in this way for the release fraction distributions, new accident sequences were thus effectively generated in each release category. A release category thus included not only the originally assigned accident sequences but also the accident sequences in the adjacent category (reduced to 10% of their predicted probability) plus the accident sequences in the next adjacent category (with the probability reduced to 1%), etc. The effect of this smoothing was to reduce even further the potential impact of common modes that had not been accounted for in the earlier, detailed analyses. This probability assignment, subdividing the distribution into order-of-magnitude areas, is depicted below.



To incorporate the probability of the sequences, lying in the category, the original sequence probability (initiating event, system failure, and containment mode) was multiplied by the appropriate factor (0.78,[1] 0.10, etc.). New sequences were thus effectively generated with the additional probability factors and were treated in the calculations as any other sequence.

An illustration of how this "smoothing" can reduce the impact of potentially undefined common mode failures can be seen by examination of Table V 3-14. The large LOCA sequence ACD-α, in release Category 1 is seen to have a probability of 5×10^{-11}. In the adjacent Category 2, there exists an interfacing systems LOCA sequence V (check valve) having a probability of 4×10^{-6}. The application of smoothing generates new sequences located in categories adjacent to Category 2 as indicated in the table. One such sequence is in Category 1 with a probability that is 10% of the V sequence in Category 2. This generated sequence probability dominates the ACD-α probability of 5×10^{-11} by about 4 orders of magnitude.

Because of the much higher probability of the generated sequence, potential common modes that may not have been discovered in ACD-α, would have to increase the ACD-α probability by four orders of magnitude. In view of the detailed common mode analyses and sensitivity studies described in Appendix III and the Addendum to the Main Report, it would not be expected that common mode contributions of that size had been overlooked.

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[1] In application, because of the minor change compared to the uncertainties, the factor of 0.78 was approximated by unity (1.0).

Figure V 4-1 also illustrates the result of smoothing by the above probability technique as applied to the small LOCA sequences for the PWR. The figure is representative of the types of result that were obtained and can be taken here as a general illustration. In the figure, the histograms correspond to various release categories.

4.1.3 ERROR PROPAGATIONS

To take into account the uncertainties in the computed probabilities of the various release categories, the error spreads for each of the variables in the accident sequence were propagated in the same manner as had been previously done for the individual fault trees. The sequences generated by the smoothing technique to account for release variations discussed in the previous section were simply treated as additional sequences. The error spreads on the initiating event probabilities and containment failure mode probabilities came directly from the data assessments. The error spreads on the system failure probabilities came from the Monte Carlo calculations which had been performed by the SAMPLE code.[1]

Given the error spreads for each of these input variables, the SAMPLE code was again run to obtain the error spread and median value applicable to the probability of a particular release category. As in the other error calculations performed in the study, log normal distributions were used for the input variables. For the system probability, a log normal was used to approximate the histogram distribution which had been calculated by the SAMPLE run (the computed system median was used with an error factor equal to the largest of the error factors associated with the upper and lower 90% SAMPLE bounds). Sensitivity studies showed the approximation to have negligible error on the results. The mechanics of the propagation were performed in exactly the same way as for the fault trees which are fully described in Appendix II.[2]

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[1] See Appendix II for a description of the SAMPLE code.

[2] In summing the event sequence probabilities for a release category, the same initiating event, system failure, etc., which occurred in several sequences needed, of course, to be treated as the same variable.

4.2 QUANTIFICATION OF LOCA EVENT TREES

The accident sequences for the PWR and BWR Large and Small LOCA event trees were quantified using input data from Appendix II - Fault Trees, Appendix III - Failure Data, and Containment Failure Mode probabilities contained in Tables 2 and 9 of attachment 1 to this Appendix. The methodology used to quantify these accident sequences is presented in the preceding section (4.1). The probabilities of the dominant (i.e., most probable) accident sequences for each event tree are presented in Tables V 3-4 to V 3-6 and V 3-9 to V 3-11.

Until now one of the headings in the large LOCA event trees has not been discussed. This is emergency core cooling functionability (ECF). This will now be discussed below.

From the beginning of the study it was recognized that considerable controversy existed about the ability of emergency core cooling systems (ECCS) to perform adequately in the event of a LOCA. The lengthy ECCS rulemaking hearing, the AEC's decision on ECCS criteria following the hearing, and the additional analytical and experimental knowledge gained since the beginning of the controversy all tend to confirm that a large degree of conservatism exists in the design of ECCS systems. Some realistic, or best estimate, calculations, as opposed to the conservative one required by the AEC rule, indicate there can be as much as 1000°F between expected temperatures and those predicted by following the AEC's criteria and calculational models (Ref. 1).

On the other hand, there are factors that could be involved in LOCAs that can potentially affect ECCS performance and that are not included in AEC's calculational models. In large LOCAs, substantial blowdown forces can occur which have the potential to damage portions of the core or RCS in ways that can affect the ability of the ECCS to adequately cool the core. For this reason, the large LOCA event tree, in addition to including considerations pertaining to operability of ECCS (ECI heading), also includes the ECCS functionability considerations (ECF heading). ECF was not included in the small LOCA event tree because, with small breaks, the blowdown forces would be very much smaller than in the large LOCA and the possibility of failures that could affect ECF is considered negligibly small.

The AEC has recognized the need to prevent blowdown forces from causing failures that could adversely affect the ECCS capability. Thus design requirements have been specified to ensure that elements within the RCS whose failure could adversely affect ECCS performance can withstand the large LOCA blowdown forces. Examples of the elements whose failure might affect the effectiveness of ECCS after the large LOCA are:

a. PWR

1. Excessive fluid leakage from steam generator to the RCS due to strucutural failures of steam generator tubes or tube sheet.

2. Excessive core bypass flow due to structural failures of core shroud or core supporting structures.

3. Excessive core distortion and flow blockage from structurally failed mechanical parts in RCS.

b. BWR

1. Excessive bypass flow due to structural failures of core shroud or core supporting structures.

2. Excessive bypass flow and inability to reflood due to jet pump structural failures.

3. Excessive flow blockage and core distortion due to structural failures of steam separators in RCS.

In recognition of the fact that it was very difficult to make a rigorous assessment of the likelihood of ECF failure, estimates of this likelihood were made on an engineering judgment basis. These estimates varied from about 10^{-5} to 10^{-2} per occurrence for large LOCA. The ECF failure values for the PWR tended to be somewhat larger than for the BWR because of uncertainties in regard to the likelihood of steam generator tube failures. It should be noted that even with values as high as 10^{-2}, ECF contributions to the overall risk assessment would be negligible.

Because of the lack of rigor associated with the above estimates of ECF failure, it was decided that the sensitivity of the large LOCA probability results would be examined, assuming various values of ECF failure probability in order to determine the potential impact on the overall risk assessment being performed. These sensitivity results for both the PWR and BWR are summarized below. They reveal that the overall results of the risk assessment would not be particularly sensitive to a wide range of ECF failure probabilities and indicated that even if values as high as 10^{-1} for ECF failure were to be used any contribution made would be within the accuracy of the overall calculations.

a. PWR

1. The large LOCA event without inclusion[1] of the probability of ECF occurrence contributed about 10% or less across the whole spectrum of releases. For that portion of the release spectrum where the larger health and property damage effects could occur (PWR release categories #1 through #4), the large LOCA event contributes less than about 5% to the overall risk assessment. These contributions can be seen in Table V-16, section 3 of this Appendix.

2. The above large LOCA contributions to the overall risk assessment could potentially increase with inclusion of ECF; however, no appreciable change would occur unless very large values for the ECF likelihood were to exist. For example if an ECF failure value as large as say 10^{-1} was applied, the results would be that the contributions of the large LOCA to the overall risk assessment would be about 30% or less at the low release end of the spectrum and less than about 10% at the high release end of the spectrum. Such values are clearly within the errors involved in the overall calculations of risk and suggest that seeking a high level of confidence on the likelihood of ECF failure was not necessary for purposes of this study.

b. BWR

1. The large LOCA event without ECF probability values in-

[1]Note that in Tables V 3-4 and V 3-9 the large LOCA sequences potentially involving ECF were carried along in the respective release categories. These sequences were not included in the listing of dominant LOCA sequences.

cluded contributed about 1% or less across the whole spectrum of consequences as can also be seen in Table V 3-16.

2. The above result for the BWR would also be relatively unaffected for large values of ECF. For example, if an ECF value as large as 10^{-1} were to be applied, the result would be that the large LOCA contributions would increase to less than 15% over the entire release spectrum. Similar to the PWR results above, a high level of confidence regarding the likelihood of ECF failure was not required for purposes of this risk study.

4.3 QUANTIFICATION OF TRANSIENT EVENT TREES

As discussed in section 4.3 of Appendix I, a portion of the study's effort was focused on assessing risk contributions from transient events. It was demonstrated in Appendix I that this need involve only an examination of anticipated transients and that less likely transients do not contribute to accident risk.

The transient trees presented in sections 4.3.1 and 4.3.2 of Appendix I identified the systems which can affect the course of events after an initiating transient. Where they were available, system failure probabilities estimated by the fault trees presented in Appendix II were used in quantifying the transient tree accident sequences. In addition, data available from reactor operating experience were used to estimate the unavailability of systems (such as the main feedwater system). Summaries of the PWR and BWR system failure probabilities applicable to transient event trees are presented in Tables V 4-1 and V 4-2, respectively. The probabilities of the dominant accident sequences from these trees were previously presented in Tables V 3-14 and V 3-16.

The following two sections will present the values of failure probability for each of the headings in the PWR and BWR trees, respectively. The discussion below presents the values of the transient event (T) for both trees. Each tree was analyzed as appropriate to identify the particular transient events which, along with the other system failures, contributed significantly to the risk assessment.

In quantifying (T), the first step involved examination of applicable data from nuclear power plant operating experience for 1972 (Ref. 2). This indicated a total of about 10 shutdowns per reactor year, of which 7 were due to equipment malfunctions, operator errors, etc., and caused rapid shutdown by means of the reactor protection system (RPS). Three of the ten shutdowns were orderly, slow shutdowns for such items as leaks, maintenance, etc. Of the seven RPS trips, 3 per year were due to interruptions of main feedwater and included about 2 per 10 years that were due to loss of off-site power. Based on the above data, a median value of 10 was used for T, with an error bound of 2 to cover a variation between 5 and 20 transients per year.

Those transients involving loss of main feedwater and loss of off-site power are of particular interest. The loss of main feedwater increases dependence on backup systems for removal of core decay heat in a shutdown. The loss of off-site power causes a loss of main feedwater and can potentially affect the availability of back-up heat removal systems. All the above transient events were included in the quantification of the PWR and BWR transient event trees.

4.3.1 PWR TRANSIENT TREE QUANTIFICATION

This section will present the quantification of the various events, except for T (the transient events), which have been discussed above. Table V 4-16, already presented, summarizes the probability values used in this analysis. The material below presents a discussion of the rationale for the selection of information in that table.

Reactor Protection System (K)

The median failure probability used for the reactor protection system (RPS) which serves to trip the reactor control rods and terminate core power was 3.6×10^{-5} with an error spread of about 3 based on fault tree analyses presented in Appendix II. Use of this probability value for RPS is considered to be somewhat conservative in those transient events which result from loss of off-site AC power since the power loss would be expected to interrupt holding power for the rods, causing the control rods to drop into the core. For this particular transient event, the use of this

conservatism did not lead to dominant core melt sequences.

Secondary Steam Relief and Power Conversion System (M)

This column heading represents portions of the power conversion system that provides for main feedwater delivery to the steam generators as described in some detail in section 4.3.1 of Appendix I. As noted earlier, operating experience indicates that the main feedwater (MFW) delivery can be interrupted approximately three times per year. The probability of recovery of the main feedwater system following its interruption depends on the initiating fault and the time window available to restore the system to operation. The time available to restore main feedwater delivery depends on whether or not other systems such as the auxiliary feedwater system (AFWS) operated or whether or not the RPS tripped. If, for example, the RPS fails to operate following interruption of MFW delivery, high RCS pressure levels could be reached in a few minutes, and the likelihood of recovery of main feedwater in this period is very small. On the other hand, if RPS operates following this event but the AFWS fails to operate, the time period available for recovery of MFW would be about 1/2 to 1 hour prior to boiling off the water inventory of the steam generators and loss of core heat removal capability. In order to cover both these cases, the following values were used for the probability of non-recovery of main feedwater following its interruption:

	∿1/2 to 1 hour	Several minutes
Three feedwater shutdowns per year	10^{-2} (10)	1
Loss of off-site AC power at 0.2 events per year	2×10^{-1} (3)	1

Secondary Steam Relief and Auxiliary Feedwater System (L)

The failure probability for the auxiliary feedwater system was developed by fault tree analyses which are presented in Appendix II. The failure probabilities of interest used in assessment of the PWR transient event tree were as follows:

	Probability of Failure for AFWS
For all events not including Loss of off-site AC power.	3.7×10^{-5} (error spread of about 3)
For transient event resulting from loss of off-site power.	1.5×10^{-4} (error spread of about 3)[1]

The function of secondary steam relief as discussed in section 4.3.1 of Appendix I requires operation of but several of a large number of the safety and relief valves provided in the secondary steam system. The probability of failure of one of the valves to open is about 10^{-5} per demand based on data presented in Appendix III. As will be subsequently shown in section 4.3.2, the probability of a large number of the secondary system valves' failing to operate such that the function of secondary steam relief would be lost is very small and was determined to be negligibly small when compared to the likelihood of failure of either the AFWS or the PCS.

Reactor Coolant System (RCS) Safety and Relief Valves Open (P)

The operation of the RCS safety and relief valves serves to limit RCS pressure levels and they are designed to open when the RCS pressure exceeds set pressures. As discussed in section 4.3.1 of Appendix I, analyses have indicated that the most severe potential overpressure event for the PWR is the interruption of main feedwater delivery coupled with a failure of the RPS to trip the reactor control rods. For this situation, modest RCS overpressures occur even with operation of all pressurizer safety and relief valves. Failure of one or more of the three pressurizer safety valves could however significantly increase the RCS overpressure and thus increase the likelihood of an RCS rupture. Accord-

[1] It should be noted that this number was derived from Appendix II considering that the emergency on-site AC power source would have diesel loads greatly reduced for the transient event. Compared to the LOCA events, the diesel emergency loads are about halved. Thus a factor of approximately 3 was credited to the availability of the emergency diesel generators for the transient.

ingly, the failure of one of the three pressurizer safety valves to open was deemed to be a failure in this event. A probability of 3×10^{-5}, with an error spread of about 3, was applied for failure of those valves to open.

Reactor Coolant System (RCS) Relief and Safety Valves Fail to Close (Q)

If the relief and safety valves fail to close when the RCS pressure level returns to below the valve set pressure, the RCS could depressurize. In the PWR, if the valves fail to reclose, they provide a path for coolant loss (~1/2" diameter), causing a small RCS LOCA; thus the core cooling and containment ESFs would be utilized.[1] Operating data for the PWR have shown such a failure of the RCS safety and relief valves to reclose following a transient event. Accordingly, the failure probability for the PWR safety and relief valves to reclose, based on PWR reactor operating experience, was estimated to be about 10^{-2} with an error spread of 10.

Chemical Volume and Control System (U)

As briefly described in section 4.3.1 of Appendix I, the chemical volume and control system (CVCS) is used in normal plant operation for purposes of controlling the RCS coolant volume and boron concentrations, and assists in cooling for the main RCS circulating pumps. The CVCS pumps also serve to deliver concentrated boron and emergency coolant to the core during LOCAs and transient events which cause RCS cooldown. Therefore, the failure probability for the CVCS was taken to be failure of the CVCS in the HPIS mode of delivery. The probability value used was 8.6×10^{-3} with an error spread of about 3 as developed by fault tree analyses presented in Appendix II.

Residual Heat Removal System (W)

Estimates on the availability of the RHRS were not made for the present analyses for the reasons discussed in section 4.3.1 of Appendix I. The RHRS was shown on the PWR transient tree principally for completeness. Its use depends on the successful operation of CVCS in conjunction with either the operation of main feedwater system or the auxiliary feedwater system. However, it is only used at cold shutdown and both MFW and AFWS could serve as backup heat removal systems.

Additional Considerations

In considering the PWR transient event resulting from the loss of off-site AC power, a sequence to core melt (TML) was found that had an important probability contribution across the entire release spectrum.[1] This sequence represented total loss of all feedwater (main and auxiliary) and thus represented a loss of both normal and alternate plant heat removal systems. If both the main feedwater and the auxiliary feedwater systems fail to operate following this transient, then the steam generators would be emptied within about 1 hour. The discharge of RCS coolant through the pressurizer safety and relief valves (which would be caused by the loss of plant heat removal) would result in the eventual uncovering of the reactor core. Within about 2 to 3 hours core melt would be underway. Containment ESFs could mitigate the release of radioactivity in this core melt sequence; however, the availability of the containment ESFs and their usefulness would be conditional on recovery of AC power within this time period. The overall elements of probability that were associated with the TML sequence, including the availability of containment ESFs and the containment failure modes, are illustrated in Fig. V 4-2 through the use of a simplified event tree.

Consider, as an example from the simplified event tree above, the sequence $TMLB'-\varepsilon$, where B' represents the probability of non-recovery of off-site and on-site AC power in about 3 hours if they both failed. This sequence was found to be one of the more important ones, and the elements of probability are:

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[1] It should be noted here that for select PWR transient sequences that involve failure of the reactor protection system (RPS) to trip the reactor control rods and a failure of the RCS safety/relief valves to reclose, the PWR small LOCA event trees were considered to be applicable. However, the small LOCA trees show that, when the reactor protection system fails to operate, a core melt was considered to occur. This decision on the small LOCA trees was made because core temperature levels could potentially become unacceptably high if RPS fails. The same core melt decision was made regarding the applicable transient event sequences. This is believed to be a conservative decision.

[1] See Table V 3-14.

$$P_{TMLB'-\varepsilon} = \overset{T}{P_1} x \overset{M}{P_2} x \overset{L}{P_3} x \overset{B'}{P_4} x P_5 x \overset{\varepsilon}{P_6} \approx 6 \times 10^{-7}$$

where:

P_1 = Loss of off-site AC power (represents an interruption of the main feedwater delivery provided by the plant power conversion system, PCS).

P_2 = Non-recovery of off-site power in about 1/2 to 1 hour (represents loss of feedwater delivery provided by the plant power conversion system, PCS).

P_3 = Failure of auxiliary feedwater system, AFWS, (principal failures include failure of on-site emergency AC power and the failure of the steam turbine driven auxiliary feedwater pump).

P_4 = Non-recovery of off-site AC power for the containment ESFS within a period of about 1 hour up to about 3 hours following the transient event.

P_5 = Non-recovery of on-site emergency AC power for the containment ESFs within a period of about 1 hour up to about 3 hours following the transient event.

P_6 = Probability that containment eventually ruptures by the path involving a meltthrough of the containment vessel base mat.

Probability values for the above elements of sequence TMLB′-ε are based on the following table.

The remaining transient sequences, e.g., TMLB′-α, TMLB′-β, etc., were evaluated in like manner. Given that electric power is available to operate the containment ESFs, those sequences involving failure of containment ESFs, e.g., TMLC-α, TMLF-β, etc., relied on probability values derived from fault tree analyses presented in Appendix II.

4.3.2 BWR TRANSIENT TREE QUANTIFICATION

This section will present the quantification of the various events, except for transient events (T), which have been discussed in previous sections. Table V 4-2, summarizes the results of this analysis. The material below presents a discussion of the rationale for the selection of information in that table.

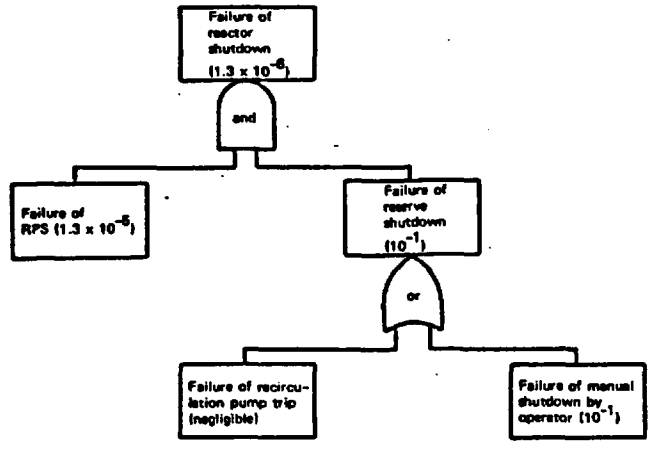
	Probability Values	Source
P_1	$\sim 2 \times 10^{-1}$	Appendix III, section 6.3
P_2	$\sim 2 \times 10^{-1}$	Fig. III-13 - Appendix III
P_3	$\sim 1.5 \times 10^{-4}$	Appendix II and Appendix III
P_4	$\sim 5 \times 10^{-1}$ (approximate ratio of the 3 hour value to the 1 hour value)	Fig. III-13 - Appendix III
P_5	≤ 1	Appendix III
P_6	≈ 0.2	Attachment 1 to this appendix

rationale for the selection of information in that table.

Reactor Shutdown (C)

Reactor shutdown can be accomplished in one of two ways, either by the reactor protection system (RPS) or by the combination of recirculation pump trip and operator actions to render the reactor subcritical. The median failure probability of the RPS is 1.3×10^{-5} with an error spread of 3 based on the fault tree analysis shown in Appendix II. The failure of the recirculation pump trip and the operator actions to render the reactor subcritical is controlled by the probability that the operator will fail to initiate the liquid poison injection

Simplified Fault Tree For Event C



system or manually initiate insertion of the control rods. Based on the analysis of operator performance in similar situations as discussed in Appendix II, and considering these factors (such as key locked switches and operating instructions) this probability has been estimated to have a median value of about 10^{-1}. The failure to achieve reactor shutdown methods is thus 1.3×10^{-6} with an error spread of 4.

Relief and Safety Valves Open (M)

The relief and safety valves are designed to open when the reactor vessel pressure exceeds set pressures. As discussed in section 4.3.2 of Appendix I for the most severe transient, 8 of the 13 safety and relief valves must open to prevent overpressure of the reactor coolant pressure boundary.

Based on the data presented in Appendix III the probability of failure of one relief or safety valve to open is less than 10^{-4}. Using the following equation:

$$P = \frac{n!}{i!\ (n-i)!} \times (10^{-4})^{i} \times (1 - 10^{-4})^{n-i}$$

the probability of two relief or safety valves failing to open is approximately 8×10^{-7}, the probability of three relief or safety valves failing to open is approximately 3×10^{-10}, etc. Therefore, even accounting for possible common mode failures, the failure probability of six safety or relief valves is so small that this event failure is insignificant when compared to the other event failures and to the overall risk assessment.

Relief and Safety Valves Fail to Close (P)

If the relief and safety valves open, they must close when the reactor vessel returns to below set pressure. If any one valve fails to close after opening, the reactor vessel will continue to depressurize resulting in a blowdown with the accompanying loss of water inventory. Based on operating data for BWR reactors, this failure probability has been estimated to be 10^{-1} with an error spread of 3.

Availability of Feedwater for Make-up Inventory (Q)

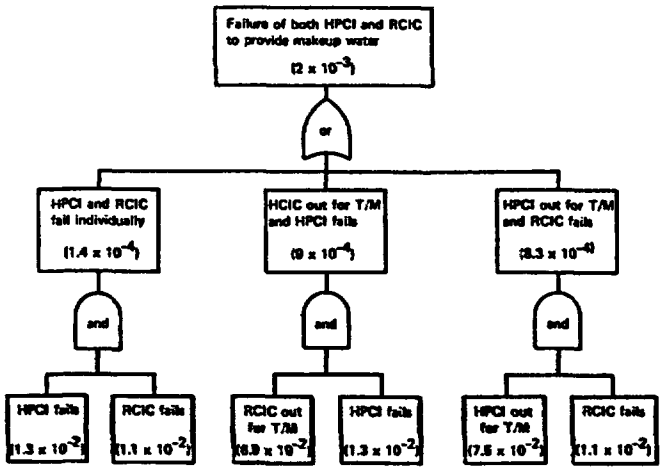
Following a transient event and other failures which cause a loss of water inventory, the normal plant feedwater system may be available. For those initiating events which involve the interruption of feedwater (approximately 3 per year), a one half hour time period exists under those conditions to restore the capability of the feedwater system to provide make-up water.

The probability of the loss of feedwater for greater than about one half hour has been estimated, based on U.S. power reactor operating experience, to be 10^{-2} with an error spread of 10.

In the event that the initiating transient is the loss of off-site power, the availability of feedwater (i.e., the electrically powered condensate pumps) is controlled by the recovery of off-site power. Based on the data presented in section 6.3 of Appendix III, the failure to recover off-site power within about one half from approximately 2×10^{-1} with an error spread of 3.

Availability of HPCI or RCIC for Make-up Inventory (U)

The unavailability of each of the high pressure coolant make-up systems (i.e., the HPCI and the RCIC) was determined from fault tree analyses as found in Appendix II. When combining these systems to obtain an overall value for the unavailability of high pressure coolant make-up systems, consideration had to be given to the fact that the plant technical specifications do not permit both systems to be out of service simultaneously for test and maintenance. This fact is taken into account on the simplified fault tree presented below.

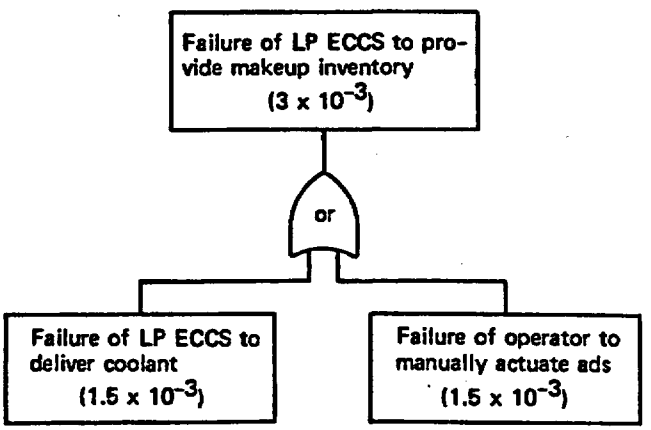


Simplified Fault Tree For Event U

The failure probabilities for HPCI (9.8×10^{-2} with an error spread of 3) and RCIC (8×10^{-2} with an error spread of 3) were developed by fault tree analysis presented in Appendix II. However, the HPCI unavailability was made up of single failure contribution (1.3×10^{-2}) and test and maintenance contribution (7.5×10^{-2}) and the RCIC unavailability was made up of single failure contribution (1.1×10^{-2}) and test and maintenance contribution (6.9×10^{-2}). The overall probability for event U was 2×10^{-3} with an error spread of about 4.

Availability of Low Pressure ECCS for Make-up Inventory (V)

The failure probability for LP ECCS (1.5×10^{-3} with an error spread of 3) was developed by fault tree analysis presented in Appendix II. The operator must actuate the automatic depressurization system (ADS) for the LP ECCS to operate. Based on the analysis of operator performance in similar situations as discussed in Appendix III, this probability has been estimated to be approximately the same as the LP ECCS failure, i.e., 1.5×10^{-3} with an error spread of 3. The failure of event V is approximately 3×10^{-3} with an error spread of 3.



Simplified Fault Tree For Event V

Removal of Decay Heat by RHR System or Power Conversion System (W)

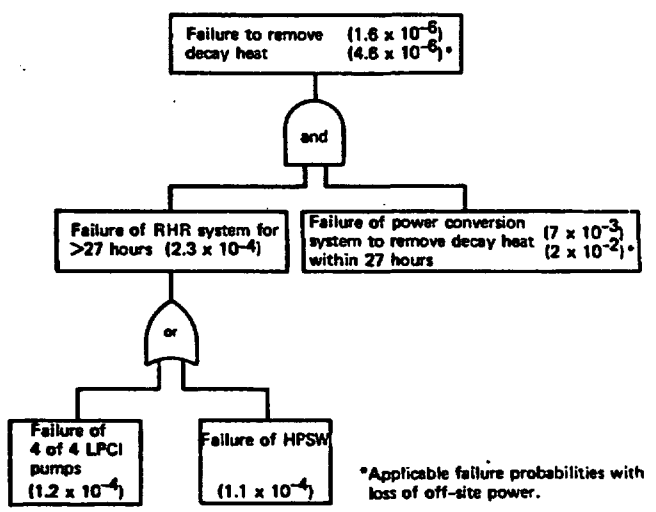
The failure of the RHR system is controlled by the failure of 4 of 4 LPCI pumps or the failure of HPSW. The failure probabilities of 4 of 4 LPCI pumps (1.2×10^{-4} with an error spread of 3) and HPSW (1.1×10^{-4} with an error spread of 3) were developed by fault tree analysis presented in Appendix II.

The failure to remove decay heat gradually builds up pressure in the containment. The data in Appendix VIII show this will result in containment overpressure in approximately 27 hours. The unavailability of the PCS to remove decay heat within 27 hours has been estimated from operating experience of U.S. power reactors to have a probability of 7×10^{-3} with an error spread of 10.

In the event that the initiating transient is the loss of off-site power, the availability of the power conversion system to remove decay heat is controlled by the recovery of off-site power. Based on the data presented in section 6.3 of Appendix III, the failure to recover off-site power within 27 hours is approximately 2×10^{-2} with an error spread of 3.

The failure probability of event W (except for loss of off-site power) is estimated to be 1.6×10^{-6} with an error spread of 10.

The failure probability of event W with the loss of off-site power as the initiating transient is estimated to be 4.6×10^{-6} with an error spread of 4.



Simplified Fault Tree For Event W

Additional Considerations

In addition to considering the above events for the transient tree, investi-

gations were made concerning the event of the total loss of AC power. This event would require the loss of all four standby diesel generators and the loss of off-site power. Given this event, the only systems which could operate would be HPCI and RCIC which provide make-up water to the reactor vessel. If make-up water is available, there is a time period of approximately 27 hours prior to core melt (see discussion of event U). If make-up water is not available, there is approximately one half hour prior to start of core melt.

The probability of core melt due to loss of AC power and loss of make-up water would be:

$$P_{CM} = P_1 \times P_2 \times P_3 \times P_4 \times P_5 \simeq 2 \times 10^{-8}$$

where:

P_1 = Loss of off-site power

P_2 = Loss of standby diesels (3 of 4)

P_3 = Loss of HPCI and RCIC

P_4 = Non-recovery of off-site power in about 1/2 hour

P_5 = Non-recovery of diesels in about 1/2 hour

The probability of core melt due to loss of AC power with make-up water available (i.e., HPCI or RCIC operate for 27 hours) would be:

$$P_{CM} = P_1 \times P_2 \times P_6 \times P_7 \simeq 10^{-7}$$

P_6 = Non-recovery of off-site power in 27 hours

P_7 = Non-recovery of diesels in 27 hours

The above values are based on the following table.

On an overall basis, it can be seen that the probability of core melt due to loss of all AC power is in the range of 10^{-7} to 10^{-8}. This probability is small compared to the dominant transient accident sequences which can lead to core melt and is not significant to the overall risk assessment.

	Probability	Data Source
P_1	~0.2	Appendix II and Appendix III
P_2	10^{-3} to 10^{-4}	Appendix II and Appendix III
P_3	~2×10^{-3}	Fault Tree Analysis - Appendix II (Event U)
P_4	~0.2	section 6.3 - Appendix III
P_5	~1	Appendix III
P_6	~2×10^{-2}	section 6.3 - Appendix III
P_7	~0.1	Appendix III

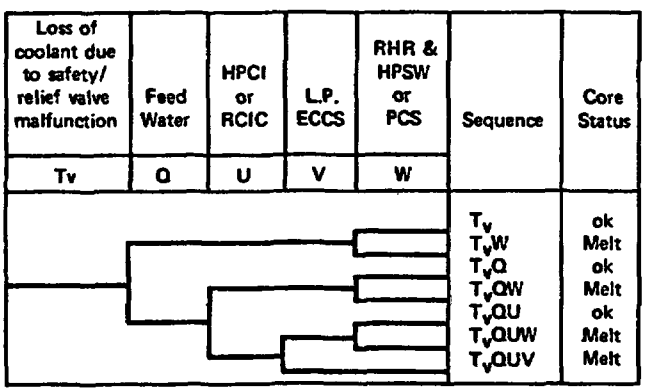
Consideration was also given to the inadvertent release of reactor coolant from the primary system pressure boundary due to actuations of the safety/relief valves and the failure of the valves to reclose after actuation. Operating experience from BWRs has indicated that such valve failures have occurred at a frequency of about 0.1 per plant per demand for operation of the safety/relief valves. In general, these incidents have led to depressurization of the primary system and have either caused or have been the result of transients requiring the plant to be shutdown. For these reasons such valve events were incorporated into the BWR event tree for the anticipated transients.[1] It can also be seen from operating experience that the coolant loss in general, has been of small magnitude which has not led to automatic actuation of the emergency core cooling systems because plant makeup systems normally in use (such as feedwater/condensate pumps and the control rod drive pumps) have provided for the resultant losses in coolant inventory through the failed-open valves. Thus these incidents would not be classed as LOCAs. Nontheless, the frequency of these incidents has been of some concern to safety authorities (Ref. 3).

To provide context for these safety/relief valve incidents as they may pertain to core melt probability, a simplified event tree was used as shown

[1] Section 4.3.2 of Appendix I.

below. The initiating event (T_V) was postulated to be the failure of the safety/relief valves to close after being opened. The other event headings shown on the simplified tree are consistent with those used in connection with evaluation of the more generalized BWR transient Event Tree (Fig. I 4-16 of Appendix I). The median value selected to describe the anticipated frequency for the initiating event, T_V, was 1.0 per reactor year, i.e.

$$\left(\sim 10 \frac{\text{valve demands}}{\text{reactor year}} \times \sim 0.1 \frac{\text{valve malfunctions}}{\text{demand}}\right).$$



The sequences which can potentially lead to a core melt i.e., T_VW, T_VQW, T_VQUW, and T_VQUV, are found to be at least an order of magnitude lower than previously identified core melt paths in the transient event tree of Appendix I. Thus BWR incidents involving such depressurizations then do not appear to have a significant impact on the BWR core melt probability.

4.4 QUANTIFICATION OF PWR CHECK VALVE EVENT TREE

Section 4.1.5 of Appendix I summarized the investigations made to determine if pathways existed in the PWR plant whereby an RCS LOCA could potentially bypass the containment ESF's and contribute importantly to the overall risk. As noted in section 4.1.5, one situation involving the LPIS was identified. The failure of check valves in the LPIS was determined to have an important probability contribution to the high consequence (high radioactive release magnitude) portion of the consequence spectrum. Estimates of the probability for LPIS check valve failure are presented below. Figure V 4-3 illustrates that portion of the LPIS design of interest. The probabilities of the dominant accident sequence for the LPIS check valve event tree has previously been shown in Table V 3-8 part B.

4.4.1 PROBABILITY OF VALVE FAILURE

-Let the initiating event V be V_1 failed open and then V_2 rupturing:[1]

-Let λ_1 be the failure rate of V_1 and λ_2 be the failure rate for V_2:

-Let Q be the probability of V occurring in time for any one leg:

$$Q = \int_0^t \lambda_1 dt' \int_{t'}^t$$

$$\lambda_2 dt'' = 1/2\ \lambda_1 \lambda_2 t^2$$

-Q_{sum} for the three legs is:

$$Q_{sum} = 3Q = 3/2\ \lambda_1\ \lambda_2\ t^2$$

-Q_{sum} is the probability of V_1 failing open then V_2 rupturing in any of the three legs.

-From Appendix II data base:

$$\lambda_1 = 3 \times 10^{-7}/\text{hr}$$
$$= 2.6 \times 10^{-3}/\text{year (3), open}$$

$$\lambda_2 = 1 \times 10^{-8}/\text{hr}$$
$$= 8.8 \times 10^{-5}/\text{year (10), rupture}$$

$$Q_{sum} = 3.4 \times 10^{-7}\ t^2,\ t \text{ in years}$$

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[1] Rupture means failure of check valve disk which allows the RCS to communicate with the LPIS.

Averaging Q_{sum} over, say, 5 years of plant operation, which corresponds to about 2 years of operation without corrective design action:

$$Q_{sum}^{(1)} = 3.4 \times 10^{-7}$$
$$Q_{sum}^{(2)} = 1.4 \times 10^{-6}$$
$$Q_{sum}^{(3)} = 3.1 \times 10^{-6}$$
$$Q_{sum}^{(4)} = 5.4 \times 10^{-6}$$
$$Q_{sum}^{(5)} = 8.5 \times 10^{-6}$$

Average Q_{sum} per year is found to be approximately:

$$\bar{Q}_{sum} \approx 2 \times 10^{-6}/\text{year} \ (\sim 10)$$

Another possibility exists that V_2 fails open and V_1 ruptures in time t, in which case,

$$Q_{sum} = 2(3Q) = 3\lambda_1\lambda_2 t_2$$

(i.e., the probability of V occurring in time t is doubled). The average $\bar{Q}_{sum}$ would thus become:

$$\bar{Q}_{sum} = 4 \times 10^{-6}/\text{year} \ (\sim 10),$$

where the error factor is estimated to be approximately 10.

Effect of Testing Check Valves

If the LPIS check valves were to be periodically tested to determine their status as barriers, a decrease of the average failure probability for the check valves would occur. The magnitude of improvement would depend on the freqency of the periodic testing. If, as an example, the check valves remain undisturbed during plant operation except for the once-per-year flow testing for the LPIS and the check valve barrier status was determined yearly after such disturbance, then the failure probability would be about 6.8×10^{-7}. (This is the yearly rate as shown above; i.e.,

$$2\,Q_{sum}^{(1)} = 2(3.4 \times 10^{-7})$$
$$= 6.8 \times 10^{-7}/\text{year}).$$

This indicates that yearly testing could yield about an order of magnitude decrease compared to the average probability of failure for the LPIS check valves. If more frequent testing was to be performed, there would be a further decrease in the failure probability in accord with the following:

$$Q_{sum}\ (\text{repair}) = \frac{\tau}{t}\, Q_{sum}.$$

For example, if monthly testing was to be performed (i.e., $\tau = 1/12$ year), the failure probability would be

$$Q_{sum}\ (\text{repair}) = 1/12\ Q_{sum}$$
$$= 1/12\ (6.8 \times 10^{-7})$$
$$\cong 6 \times 10^{-8}/\text{year}.$$

In summary, at least an order of magnitude decrease in the failure probability for the LPIS check valves can be obtained by a reasonable testing program.

4.5 QUANTIFICATION OF REACTOR VESSEL RUPTURE ACCIDENTS

The failure rate for reactor vessel rupture is based principally on the Advisory Committee on Reactor Safeguards (ACRS) Report, "Integrity of Reactor Vessels for Light Water Power Reactors", dated January 1974.

Since the experience with nuclear pressure vessels was too limited to permit relevant statistical inferences of these failure probabilities, the ACRS report was based on data of many types of non-nuclear pressure vessel failures available from the United Kingdom and Federal Republic of Germany, as well as data from the United States, such as Edison Electric Institute - Tennessee Valley Authority data; Edison Electric Institute Boiler Drum and Pressure Vessel data; American Boiler Manufactors

Association data; United States Navy experience; and United States commercial reactor experience. Many of these data pertain to vessels used in power plant applications.

The following policy was followed by the ACRS in deriving failure rates and probabilities:

a. Failure rates quoted in the literature are given as stated;

b. Other failure rates are calculated simply by dividing the number of failures by the number of vessel-years;

c. Where possible in items a and b, the 99 percent confidence upper bound failure probability is also given.

The summary of section 5, "Pressure Vessel Failure Statistics and Failure Probabilities," of the ACRS report states:

"The Committee concludes that there is reasonable assurance that: (1) the disruptive failure probability of non-nuclear vessels in central station service by modes pertinent to reactor vessels is less than 1×10^{-5} per vessel-year, (2) the disruptive failure probability of reactor vessels designed, constructed, and operated to Sections III and XI of the Code is less than 1×10^{-6} per vessel-year, and, (3) the disruptive failure probability of such reactor vessels, beyond the capability of engineered safety features is even lower."

To arrive at an estimate for a median value and error bounds for reactor vessel rupture used in this study, two considerations noted in the ACRS study are significant. One is that the disruptive reactor vessel failure probability is less than 10^{-6} per vessel-year. The second consideration is that the likelihood of reactor vessel ruptures beyond the capability of engineered safety features is even lower. The safety study's analysis and review generally agreed with these results and a value of 10^{-7} was used for ruptures of the reactor vessel large enough to be beyond the capability of ECC systems. An error spread of a factor of 10 was associated with this value which gave an upper bound of 10^{-6}, coinciding with the ACRS value for any rupture.

The methodology described in section 4.1 of this appendix was used to quantify the PWR reactor vessel rupture event tree (Appendix I, section 4.1.4) using the above information for the failure rate of the initiating event.

The above information for the failure rate of the initiating event was also used for the special case of concern in a BWR reactor vessel rupture (Appendix I, section 4.2.5), i.e., a PV rupture that creates primary containment rupture in such a way that an air oxidation path is created. The conditional probability that a primary containment rupture would create an air oxidation path was estimated at 10^{-1} with an error spread of 3. The PWR and BWR vessel rupture accidents sequences with their associated probabilities were assigned to the appropriate release categories described in section 3 of this Appendix and are presented in Table V 3-14 and V 3-15.

4.6 DESCRIPTION OF DOMINANT ACCIDENT SEQUENCES

Sections 4.6.1 and 4.6.2 that follow present a summary description of the accident events and sequences that were found to dominate the probability of the various PWR and BWR release categories shown in Tables V 3-14 and V 3-15. Only those release categories that involve core melting are described (e.g., #1 through #7 for the PWR and #1 through #5 for the BWR), and within each of these categories the sequences that dominate the probability of occurrence of each category, down to those that contribute as little as 10%, are described.

4.6.1 PWR DOMINANT ACCIDENT SEQUENCES

The probability contributions that dominated the spectrum of PWR releases were found to be shared largely by three accident events. As can be readily seen by Table V 3-14, these events were:

1. The PWR small LOCA,

2. The PWR transient event involving loss of offsite power,

3. The PWR LPIS check valves.

Only a small contribution was made by the PWR large LOCA event and essentially no contribution was made by the reactor

vessel rupture event.[1] For ease of understanding the descriptions of the dominant sequences to follow, the event tree headings appearing in these dominant sequences have been identified previously in Table V 3-15.

PWR Release Category 1

The dominant contributing sequence appearing in this category is a transient accident sequence TMLB'-α which is briefly described below.[2] (A small LOCA sequence S_2C is also a contribution and this is discussed subsequently in connection with PWR release category #3.)

Sequence TMLB'-α - Failure of the Feedwater Delivery Systems (Power Conversion and Auxiliary Feedwater Systems) Given the Initiating Transient Event of Loss of Offsite AC Power with a Failure to Recover Either Onsite or Offsite AC Power Within About 3 hours.

a. Status of Systems. This sequence considers that a transient event involving loss of offsite AC power occurs and that neither the power conversion system nor the auxiliary feedwater system is available to remove heat from the RCS. In determining the probability of the sequence, it was considered that neither the off-site nor the on-site emergency sources of AC power were recovered within about 3 hours, as previously detailed in section 4.3.1 of this Appendix.

b. Core Condition. Since all AC power was lost and not recovered in sufficient time to prevent an excessive coolant loss through the RCS safety and relief valves, a core melt could occur. Also, since all AC power sources were not recovered, containment ESFs could not operate to mitigate the radioactivity released from the melting core.

c. Containment Failure Mode. This sequence considers that a steam explosion (α) could occur as the molten core would drop to the lower head of the reactor vessel and contact residual vessel coolant. The steam explosion could cause a rupture of the reactor vessel and large missile penetration of the containment while the containment would be at elevated pressure. During the explosion, the volatile fission products would be finely dispersed and air oxidation would enhance the magnitude of the radioactivity release from the violated containment.

PWR Release Category 2

The dominant contributor to this category, is event V although the TMLB' sequences are nearly equal in contribution. Event V is described below.

Sequence V - Rupture of the LPIS check valves

a. Status of Systems. This event is discussed in detail in section 4.1.5 of Appendix I. In summary, failure of the LPIS check valves would result in (1) a LOCA, (2) a failure of the LPIS with resultant core melt and (3) a discharge of radioactivity which would bypass containment and would go to the atmosphere outside of containment.

b. Core Condition. Since a LOCA would occur without the required emergency core cooling capability available from the low pressure injection system, a core meltdown could result.

c. Containment Failure Modes. Since the failure of the LPIS check valves would permit release of radioactivity directly outside of containment, the status of containment would not be of immediate significance. Although event V would represent the overwhelming path for gaseous radio-

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[1] It is interesting to note that the values in Table V 3-14 for the reactor vessel contribution to risk is based on an extrapolation of a data base for non-nuclear vessels that have a median failure value of about 10^{-6}. The reactor vessel median value of 10^{-7} used herein is based on giving credit for improved techniques applicable to nuclear vessels. However, even if 10^{-6} had been used, the reactor vessel failure sequences would still have made no significant contribution to risk in either the PWR or BWR case.

[2] It can be observed from Table V 3-14 that the LPIS check valve rupture event V can have an important probability contribution to Category 1, particularly when probability smoothing between the release categories is done.

activity release from containment to the atmosphere, a small increment of release could occur from containment should the molten core undergo a steam explosion (α). Should the core eventually melt through the containment vessel base mat (ε), the increment of release would be very small and negligible compared to path V. Analysis of the radioactivity releases considering these other containment failure mode possibilities indicated that event V itself suitably represented the probability and magnitude of radioactivity released to the atmosphere during the event.

PWR Release Category 3

The dominant contributor to this category is sequence S_2C-δ. It is discussed below.

Sequence S_2C-δ - Failure of the Containment Spray Injection System Given a Small LOCA (1/2" ≤ D ≤ 2")

a. Status of Systems. This sequence assumes a small LOCA would occur having an equivalent break diameter between about 1/2 inch and 2 inches, and the containment spray injection system (CSIS) would fail to operate. Failure of the CSIS for a LOCA in this size range could result in insufficient water in the containment sump at the time the CSRS is initiated. Thus, there would be insufficient water supply for the recirculating spray pumps, and they could fail. This would result in loss of containment heat removal and the containment would fail by overpressure. Immediately following the overpressure failure of containment, the emergency core cooling system, which was considered to be satisfactorily operating in the interim, would be assumed to fail due to cavitation of the ECCS pumps.

b. Core Condition. The core would initially be cooled by ECI operation. Failure of the emergency core cooling system in the recirculation mode of operation would be due to pump cavitation and would result in core melt approximately one to two hours after the occurrence of the containment overpressure failure.

c. Containment Failure Mode. Failure of CSRS due to lack of water supply as discussed above would lead to containment overpressurization and its ultimate failure.

PWR Release Category 4

Sequences contributing to this category are sequences S_1CD-β, ACD-β and S_2DG-β. All these sequences result from either a large or small LOCA coupled with a failure of the emergency core cooling injection system and failure of one of the containment ESFs. Each is discussed below. Note that this is the first consequence category for which any sequence from the large LOCA tree becomes a contributor of any significance.

Sequence S_1CD-β - Failure of the Containment Spray Injection System and the Emergency Core Cooling Injection System Given a Small LOCA (2" ≤ D ≤ 6")

a. Status of Systems. For this sequence the containment spray injection system and the emergency core cooling injection system would fail following the occurrence of a small LOCA having a break size with an equivalent diameter between 2 inches and 6 inches. Failure of the emergency core cooling system to inject would result in core melt. Since core melt would be underway in approximately 1/2 hour, operation of the emergency core cooling system in the recirculation mode (ECR) would not preclude core melt.

The containment spray recirculation system (CSRS) would operate successfully, taking water from the sump, which would collect it from the primary system blowdown, and would spray it into the containment atmosphere. The spray water would be cooled by circulation through the CHRS heat exchangers. Sodium hydroxide addition (SHA) would be possible to enhance radioactivity removal by the spray. Partial operation of either the CSI or ECI systems pumps would provide the SHA into the containment. Thus, the CSRS would operate to control containment pressure and to remove airborne radioactivity released by the core melt.

b. Core Condition. Because of the failure of the emergency core cooling system in the injection mode, core melt would be underway in approximately 1/2 hour.

c. Containment Failure Mode. This sequence assumes that the accident occurs as described above, and it would be coupled with a failure to

properly isolate containment openings and penetrations, giving a containment leak rate equivalent to about a 3 to 4" diameter hole.

Sequence ACD-β - Failure of the Containment Spray Injection System and the Emergency Core Cooling Injection System Given a Large LOCA

a. Status of Systems. This sequence is very similar to sequence $S_1CD-\beta$, above (4a) except that the initiating event would be the large LOCA having an equivalent break size larger than six inches. As in sequence $S_1CD-\beta$, the containment spray injection system and the emergency core cooling injection system would fail to operate. The containment spray recirculation system and the containment heat removal system would operate to remove heat from the containment, and, as noted previously, sodium hydroxide would be available to enhance the ability of the spray to remove radioactivity.

b. Core Condition. Because of the failure of the emergency core cooling system in the injection mode, core melt would be underway in approximately 1/2 hour.

c. Containment Failure Mode. This sequence assumes that the accident occurs as described above coupled with a failure to properly isolate containment openings and penetrations.

Sequence $S_2DG-\beta$ - Failure of the Emergency Core Cooling Injection System and Containment Heat Removal System Given a Small LOCA (1/2" $\leq$D $\leq$2")

a. Status of Systems. This sequence assumes failure of emergency core cooling in the injection mode and failure of the containment heat removal system. The containment spray injection and recirculation systems would operate with sodium hydroxide available to remove airborne fission products. However, because of failure of the containment heat removal system, the recirculation spray system would not cool the containment. Steaming of the coolant would eventually occur following completion of water injection by the containment spray injection system. This steaming would provide a driving force for leakage out of the containment of the radioactivity released during the core melt.

b. Core Condition. Failure of the emergency core cooling injection system would result in core melt.

c. Containment Failure Mode. This sequence assumes that the accident occurs as described above with failure of the containment to properly isolate containment openings and penetrations.

PWR Release Category 5

The dominant contributor to this category is sequence $S_2D-\beta$. Other significant contributors are sequences $S_2H-\beta$, $S_1D-\beta$, $S_1H-\beta$, and $AD-\beta$. All these sequences involve the occurrence of either a large or small LOCA, a failure of emergency core cooling in either the injection or recirculation mode, and failure to isolate the containment. Each of the significant contributors is discussed below.

Sequence $S_2D-\beta$ - Failure of Emergency Core Cooling Injection Given a Small LOCA (1/2" $\leq$D $\leq$2")

a. Status of Systems. In this sequence, the emergency core cooling system would fail to adequately inject water to the core following a small LOCA having a break equivalent diameter between about 1/2 inch and 2 inches. All containment ESFs would operate as designed to control containment pressure and leakage and would serve to remove radioactivity airborne in the containment. Since the emergency core cooling system was considered to fail in the injection mode, its operation in the recirculation mode (ECR) would not prevent core meltdown.

b. Core Condition. Failure of emergency core cooling in the injection mode would result in a core melt.

c. Containment Failure Mode. This sequence assumes that the accident would occur as described above with a loss of containment integrity resulting from failure to adequately isolate containment openings and penetrations.

equence $S_2H-\beta$ - Failure of Emergency Core Cooling Recirculation Given a Small LOCA (1/2" $\leq$D $\leq$2")

a. Status of Systems. The emergency core cooling system would initially succeed in the injection mode but would fail to cool the core in the recirculation mode following a small LOCA with an equivalent break diameter between about 1/2 and 2 inches. All containment ESFs would operate successfully.

b. Core Condition. Failure of the emergency core cooling system in the recirculation mode would result in a core melt.

c. Containment Failure Mode. This sequence assumes that the accident would occur as described above with a loss of containment integrity resulting from failure to adequately isolate containment openings and penetrations.

Sequence $S_1D-\beta$ - Failure of Emergency Core Cooling Injection System Given a Small LOCA (2" $\leq$D $\leq$6")

a. Status of Systems. In this sequence a small LOCA occurs having an equivalent break diameter between about 2 inches and 6 inches and the emergency core cooling system would fail to adequately inject water into the reactor vessel to cool the core. All containment ESFs would operate as designed.

b. Core Condition. As discussed in Sequence $S_2D-\beta$ above, failure of the emergency core cooling system to cool the core during the injection mode of operation would result in a core melt.

c. Containment Failure Mode. The containment failure mode considered in this sequence is identical to that considered for the sequences above in consequence category 5.

Sequence $S_1H-\beta$ - Failure of Emergency Core Cooling Recirculation Given a Small LOCA (2" $\leq$D $\leq$6")

a. Status of Systems. This sequence assumes that a small LOCA occurs having an equivalent break diameter between about 2 inches and 6 inches and the emergency core cooling system would fail in the recirculation mode. All containment ESFs would operate as designed.

b. Core Condition. As in sequence $S_2H-\beta$ above (sequence 2 in category 5), failure to adequately cool the core because of failure of the emergency core cooling system in the recirculation mode would cause a core melt.

c. Containment Failure Mode. The containment failure mode in this sequence is identical to that considered in sequence $S_2H-\beta$, above.

Sequence $AD-\beta$ - Failure of Emergency Core Cooling Injection Given a Large LOCA

a. Status of Systems. This sequence assumes the emergency core cooling injection system would fail to adequately cool the core following the occurrence of a large LOCA. All containment ESFs would operate as designed. However, since failure of emergency core cooling would cause a core melt in approximately 0.5 hour after the occurrence of the LOCA, operation of the emergency core cooling system in the recirculation mode would not prevent continuation of the core melt.

b. Core Condition. As discussed in sequence $S_2D-\beta$ above (sequence 1 in category 5), failure of the emergency core cooling system in the injection mode results in a core melt.

c. Containment Failure Mode. The containment failure mode considered in this sequence would be identical to that considered in sequence $S_2D-\beta$ above.

PWR Release Category 6

The dominant sequence in this category is sequence $TMLB'-\varepsilon$. A smaller contribution is also made by sequence $S_2CD-\varepsilon$. These are discussed below.

Sequence $TMLB'-\varepsilon$ - Failure of the Feedwater Delivery Systems (Power Conversion and Auxiliary Feedwater Systems) Given the Initiating Transient Event of Loss of Offsite AC Power with a Failure to Recover Either Onsite or Offsite AC Power Within About 3 Hours.

a. Status of Systems. The status of systems would be identical to sequence TMLB'-α, discussed above for Consequence Category 1.

b. Core Condition. Core condition would be identical to sequence TMLB'-α, above (category #1).

c. Containment Failure Modes. For this sequence, containment failure would occur after the transient when the molten core, having already melted through the reactor vessel, would melt its way through the containment vessel base mat.

Sequence $S_2CD-\varepsilon$ - Failure of Emergency Core Cooling Injection and Containment Spray Injection Systems Given a Small Loca (1/2" ≤D ≤2") Invection.

a. Status of Systems. This sequence assumes that (1) the emergency core cooling system would fail in the injection mode and, (2) the containment spray injection system would experience an independent failure following a small LOCA having an equivalent break diamtere between about 1/2 and 2 inches. Failure of the emergency core cooling system in the injection mode would result in core melt and would obviate the need to consider operability of the emergency core cooling system in the recirculation mode. Failure of the containment spray injection system would negate the effectiveness of the containment spray, recirculation and containment heat removal systems. The containment spray injection system failure and its impact on the containment ESFs was discussed above in connection with the S_2C sequence in PWR release category 3. In the $S_2CD-\varepsilon$ sequence there is a possibility that, similar to the large LOCA sequence ACDF, the core would melt through the containment vessel concrete base mat.

b. Core Condition. Failure of the emergency core cooling system in the injection mode would result in a core meltdown being underway in approximately 0.5 hr. after the occurrence of the accident.

c. Containment Failure Mode. The containment failure mode considered in this sequence would be identical to that considered in sequence TMLB'-ε, above (category 6).

PWR Release Category 7

The dominant sequence in this category is $S_2D-\varepsilon$. Other sequences which are important contributors are $S_1D-\varepsilon$, $S_2H-\varepsilon$, $S_1H-\varepsilon$, AD-ε, AH-ε, TML-ε, and TKQ-ε. Note that all these sequences involve a containment vessel melt-through as the containment vessel failure mode.

With the exception of the two transient sequences, all involve the potential failure of the emergency core cooling system following the occurrence of a LOCA with all containment ESFs operating as designed.

As previously noted, the containment failure mode considered for all sequences would be caused by the molten core, which, having melted through the lower head of the vessel, would continue to melt through the containment vessel base mat. The status of systems and core condition for several of these sequences have been described earlier in PWR Consequence Category 5 where the containment failure mode under consideration was loss of containment integrity as indicated below:

Category 7 Sequence	Other Sequence with Identical Status of System and Core Condition
$S_2D-\varepsilon$	$S_2D-\beta$ (5a)
$S_1D-\varepsilon$	$S_1D-\beta$ (5c)
$S_2H-\varepsilon$	$S_2H-\beta$ (5b)
$S_1H-\varepsilon$	$S_1H-\beta$ (5d)
AD-ε	AD-β (5e)

The other sequences identified as major contributors to PWR Category 7 are discussed below.

Sequence TML-ε - Failure of the Power Conversion System and the Auxiliary Feedwater System Given a Transient Event

a. Status of Systems. This sequence would assume failure of both the normal and emergency means of adding water to the steam generators given a transient event. The reactor would be assumed to trip and the primary system safety and relief valves would operate as designed. Since no water can be added to the

steam generators, they would eventually boil dry and would be lost as a means of transferring decay heat. The RCS pressure would increase since the means of rejecting decay heat has been lost. The RCS pressure would equilibrate about the safety valve setpoint due to operation of the valves while the core would eventually become uncovered by loss of the RCS coolant which discharges via the safety valves. Electric power was assumed to be available to run the containment ESFs in this sequence.

b. Core Condition. After core uncovering would be caused by coolant boiloff of the primary system water, the core would melt.

c. Containment Failure Mode. As previously identified, the containment failure mode was the molten core melt-through of the lower head of the reactor pressure vessel and the containment vessel base mat.

Sequence TKQ-ε - Failure to Trip and Failure of the Primary System Safety Relief Valves to Reclose given a Transient Event

a. Status of Systems. For this sequence it would be assumed that a transient event (e.g., interruption of the main feedwater delivery) would occur with a failure of the RPS to trip the reactor control rods. The heat imbalance caused by the transient results in an RCS pressure increase, and the RCS safety and relief valves would open to limit the pressure rise. However, it is assumed that safety or relief valves would stick open and would fail to reclose. This would cause depressurization of the RCS with the reactor being critical. This sequence would thus be a delayed small LOCA without trip of the reactor control rods.

b. Core Condition. The emergency core cooling system was considered to be inadequate to cool the core given a small LOCA without trip. High core temperatures and core melt was considered to occur.

c. Containment Failure Mode. For this sequence it is assumed that the molten core eventually melts through the containment vessel base mat.

Sequence AH-ε - Failure of Emergency Core Cooling Recirculation Given a Large LOCA

a. Status of Systems. It is assumed that the emergency core cooling system would fail in the recirculation mode following a large LOCA. All containment ESFs would function as designed.

b. Core Condition. Loss of emergency core cooling would cause core melt. Should the emergency core cooling system fail at the time of transfer to the recirculation mode, the core would melt in approximately two hours.

c. Containment Failure Mode. The molten core would melt through the reactor vessel and would eventually melt through the containment vessel base mat approximately 20 hours following the LOCA.

4.6.2 BWR DOMINANT ACCIDENT SEQUENCES

All BWR accident sequences that dominate the core melt probability are from the BWR transient event tree (Appendix I - section 4.3.2). The initiating event (T) is defined as any transient event requiring reactor shutdown from hot operating conditions which results from an equipment malfunction or from any parameter exceeding its reactor trip value, but which is not a result of the failure of the reactor coolant system. Definitions of failures included in these dominant sequences are listed in Table V 3-15. For the readers convenience the failures involved in these sequences are identified below:

C - Failure to make the reactor subcritical by

a. rapid insertion of control rods (SCRAM), or

b. tripping of reactor coolant recirculation pumps, and successfully operating the standby liquid control system or driving in control rods not successfully inserted when the demand for reactor scram occurred.

W - Failure to remove decay heat by either (1) the RHR and HPSW, or (2) the power conversion system.

α - Failure of the primary containment due to a steam explosion which occurs when the molten core falls

into the water in the bottom of the reactor vessel.

γ - Failure of the primary containment due to overpressure with radioactive release discharged through the failed secondary containment (reactor building) where deposition of radioactivity can occur.

γ' - Failure of the primary containment due to overpressure with radioactive release direct to atmosphere.

BWR Release Category 1

Two sequences, TCα and TWα, were found to dominate the probability of Category 1. These sequences are described below:

a. Sequence TWα - Failure of Decay Heat Removal Given Event (T)

1. Status of Systems. This sequence considers the occurrence of a transient as defined earlier. Following the transient, the reactor would be made subcritical by control rod insertion or the combination of a recirculation pump trip and poison injection. The safety/relief valves would open if the reactor vessel pressure were to reach the relief valve set points and steam would be discharged into the suppression pool. Also for this sequence, the valves are assumed to reclose after the pressure has been relieved or reduced through either manual operation or self-activation of the valves. The pumps in either the high pressure or low pressure ECI system or the condensate system pumps are assumed to be operating to maintain adequate make-up water to the reactor vessel.

The potential failure to transfer decay heat to the environment by use of components of the power conversion system or by the use of the RHR and HPSW systems eventually would result in a core melt. The sequence of events would be as follows:

(a) In about 27 hours after the initiating transient, the continued discharge of steam to the suppression pool via the safety/relief valves would increase the containment pressure to about 175 psia.

(b) The containment would fail structurally.[1] It is assumed that this failure would terminate the replenishment of the reactor vessel water inventory by the feedwater and the control rod drive system due to the forces associated with the containment overpressure failure.

(c) It is also assumed that none of the systems, RCIC, HPCI, or the low pressure ECCS, would be able to pump water from the suppression pool to the reactor vessel due either to damaging forces from the containment failure or to inadequate NPSH because of the high temperature of the suppression pool water.

2. Core Condition. If the make-up water to the core were to be terminated, the water in the core region of the reactor vessel will boil off and the core will melt.

3. Containment Failure Modes. The primary containment would fail due to overpressure prior to the initiation of core melt. This, in turn, would fail the reactor building. Some of the radioactivity that could be released during the initial core melt period can be released at near ground level (i.e., at 25 meters or less). When the molten core would fall into the bottom of the reactor vessel containing water, a steam explosion and upward bound missiles can potentially occur. It is assumed that the steam explosion would also expel a portion of the molten core to the outside atmosphere and this would result in a large, elevated release of radioactivity.

b. Sequence TC-α - Failure to Make Reactor Subcritical, Given Event (T)

1. Status of Systems. This sequence considers the occurrence of a transient which necessitates a reactor shutdown from a hot operating condition. How-

[1] See Appendix VIII.

ever, the reactor may not be made subcritical due to failure to insert control rods and failure of the reserve shutdown means (i.e., principally the failure of liquid poison injection).

2. Core Condition. With the reactor not made subcritical, the reactor would tend to remain at relatively high power immediately following the transient. After steam flow to the turbine would be terminated due to the closure of the turbine stop valve or the main steam isolation valve, the reactor pressure would increase. This pressure increase would lead to a rise in power which, in turn, would further increase the primary coolant system pressure. The opening of the primary system relief and safety valves would limit the pressure increase; the initial peak pressure attained will be a function of the transient power history and the setpoints and capacities of the safety and relief valves. A peak primary system pressure of 1550 psia would be predicted (Ref. 4) for a plant of the type considered here. Recirculation pump trip combined with the loss of moderator through the relief and safety valves would tend to reduce the reactor power level. The power level would be expected to stabilize at about 30 percent of nominal. The HPCI system would start to add water to the primary system shortly after the initial pressure surge subsides. However, at power levels that are significantly above decay heating, the boiloff rate would be greater than the capacity of the HPCI; thus, the water level in the primary system would decrease and eventual core meltdown could be expected.

3. Containment Failure Mode. When the molten core would fall into the bottom of the reactor vessel containing water, a steam explosion might occur which could overpressurize and rupture the primary and secondary containments. It is assumed that the steam explosion could expel a portion of the molten core to the outside atmosphere, potentially causing a large radioactivity release similar to the TW-α sequence previously described.

BWR Release Category 2

a. Sequence TW-γ' - Failure of Decay Heat Removal Given Event (T)

1. Status of Systems. This sequence would be identical to sequence TW-α above regarding the status of systems.

2. Core Condition. This sequence would be identical to sequence TW-α above regarding the core condition except that no steam explosion occurs.

3. Containment Failure Mode. This sequence would involve the overpressure failure of containment prior to core melt similar to sequence TW-γ described below except the failure of containment would be assumed to result in a flow path directly to the environment. That is, there would be negligible deposition in the annulus between the containment and its concrete shield and by the reactor building for this TW-γ' sequence. Examination of the plant layout revealed that about a 0.2 probability existed for this leakage path for all overpressure failures of containment. The discharge of radioactivity was assumed to occur at near ground level for this γ' sequence. Also, the large magnitude of release from the containment for this category of release would be due in large part to the fact that the containment was breached prior to meltdown when the capability of the suppression pool to retain radioactivity was limited because of elevated pool temperatures.

BWR Release Category 3

a. Sequence TW-γ - Failure of Decay Heat Removal Given Event (T)

1. Status of Systems. This sequence would be identical to sequences TW-γ' and TW-α regarding the status of systems.

2. Core Condition. This sequence would be identical to sequence TWγ' and TW-α regarding the core condition except that no steam

explosion occurs during the melt process in the reactor vessel.

3. Containment Failure Mode. The containment would be assumed to fail by overpressure prior to the core melt. The overpressure failure would occur in a location where the radioactivity released from the core meltdown could deposit on the surfaces of the containment annulus space and the reactor building prior to its release to the atmosphere. It is assumed for this sequence that the so-called "blowout" panels in the reactor building would be likely to be blown away from the reactor building during the overpressure failure of the primary containment and this provides the principal leakage path for the radioactivity.

b. Sequence TC-γ - Failure to Make Reactor Subcritical Given Event (T)

1. Status of Systems. The status of systems would be identical to sequence TC-α above.

2. Core Condition. This sequence would be identical to sequence TC-α except that no steam explosion occurs.

3. Containment Failure Mode. This sequence and the leakage paths would be effectively the same as for sequence TW-γ above except that the containment failure by overpressure would be attributable to the ongoing core melt processes. This TC-γ sequence also would include the possibility of TC-γ' where the released radioactivity from containment would go directly to the environment similar to TW-γ' above. Estimates on the magnitude of releases from TC-γ' showed that the releases would be essentially identical to TC-γ due to the large retention of the radioactivity in the suppression chamber and pool.

BWR Release Category 4

The accident sequences in Category 4 would involve failure of the same basic systems as considered in the previous release categories and principally include failure of containment in ways found to be much less probable than the overpressure failure modes in Categories 2 and 3. In Category 4, the dominant sequences involve the potential failure of containment to properly isolate; this would cause initial containment leakage to be very high (>100 v/o per day) relative to the design basis leakage (<1 v/o per day).

The high initial leakage could preclude containment failure by the overpressure modes described for release Categories 2 and 3 and the radioactivity leaked from the primary containment during a core melt could be significantly reduced by operation of the reactor building standby gas treatment system and by filtration provided within this system. Category 4 includes a number of sequences which examined variations in the leakage rate, leakage locations, and the effectiveness of the standby gas treatment system. Nonetheless, the probability for core melt release of Category 4 magnitude or less was found to be dominated by the adjacent Categories 2 and 3 described previously.

References

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," USAEC, Docket RM-50-1, Dec. 28, 1973, Washington, D.C.

2. "Evaluation of Nuclear Power Plant Availability," USAEC, Jan. 1974, OOE-ES-001.

3. Evaluation of Incidents of Primary Coolant Release from Operating Boiling Water Reactors, (Report dated October 30, 1972 prepared by The AEC Office of Operations Evaluation).

4. Michelotti, L. A., "Analysis of Anticipated Transients Without Scram," General Electric, APED, NEDO-10349 (March, 1971).

TABLE V 4-1 PWR EVENT TREE PROBABILITIES FOR TRANSIENT EVENT TREE

Event	Failure Probability	Data Source
Reactor Protection System (K)	3.6×10^{-5} (4) [a]	Appendix II & III
Availability of Power Conversion System for Main Feedwater Delivery (M)		
a. ≥ 30 minutes with off-site AC Power Available	1×10^{-2} (10)	U.S. Power Reactor Operating Experience
b. For ≥ 30 minutes without off-site AC power available	2×10^{-1} (3)	Appendix III
Auxiliary Feedwater System (L)		
a. With off-site AC power available	3.7×10^{-5} (3)	
b. Without off-site AC power available	1.5×10^{-4} (3)	Appendix II
RCS Relief and Safety Valves Open (P)	3×10^{-5} (3)	Appendix III
RCS Relief Valves Reclose (Q)	$\sim 10^{-2}$ (10)	PWR reactor operating experience
CVCS HPIS Mode of Operation (U)	8.6×10^{-3} (3)	Appendix II & III
RHRS (W)	Not Required	-

(a) Number in parentheses indicates error bound, i.e., 1×10^{-2} (10) has a lower bound of 1×10^{-3} and an upper bound of 1×10^{-1}.

TABLE V 4-2 BWR EVENT FAILURE PROBABILITIES FOR TRANSIENT EVENT TREE

Event	Failure Probability	Data Source
Reactor Shutdown (C)	1.3×10^{-6} (4)	Appendix II Appendix III
Relief and safety valve open (M)	~0	Appendix III
Relief and safety valves close (P)	1×10^{-1} (3)	BWR operating experience
Feedwater availability (Q) (with off-site power)	1×10^{-2} (10)	U.S. power reactor operating experience
Feedwater availability (with loss of off-site power)	2×10^{-1} (3)	Appendix III
HPCI or RCIC availability (U)	7.8×10^{-3} (4)	Appendix II
LP ECCS availability (V)	3×10^{-3} (3)	Appendix II Appendix III
Decay heat removal (W) (with off-site power)	1.6×10^{-6} (10)	Appendix II Appendix III
Decay heat removal (W) (with loss of off-site power)	4.6×10^{-6} (4)	Appendix II Appendix III

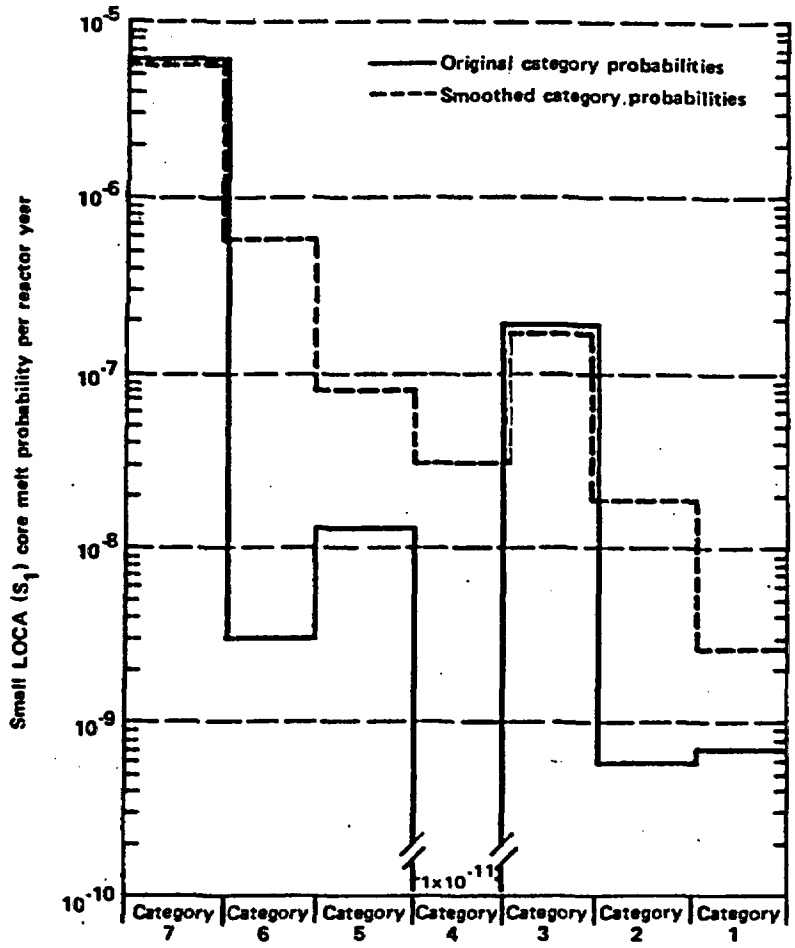

FIGURE V 4–1 APPLICATION OF PROBABILITY SMOOTHING

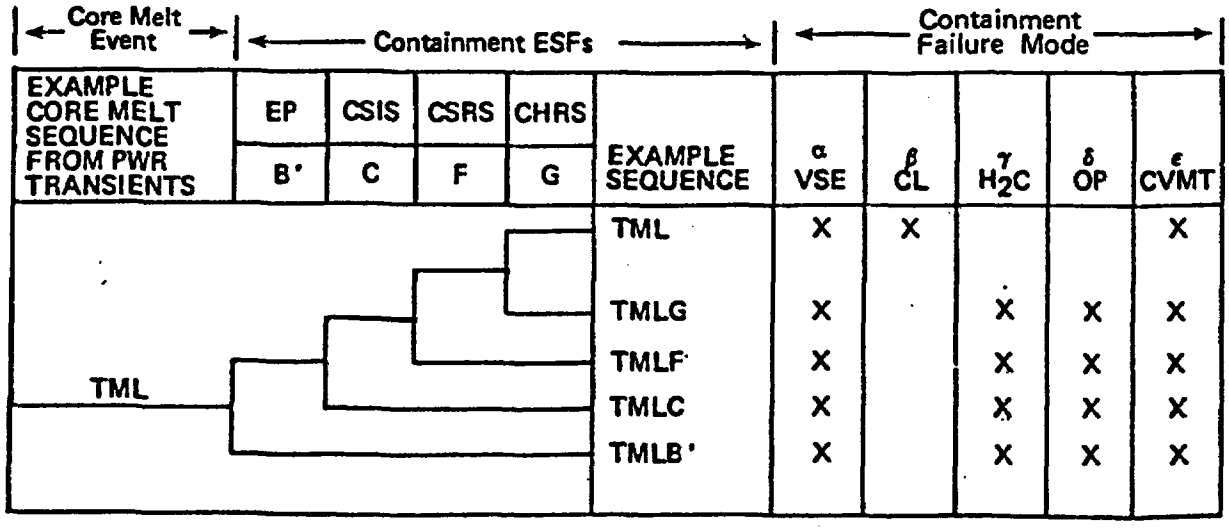

FIGURE V 4–2 SIMPLIFIED EVENT TREE FOR TRANSIENT SEQUENCES INVOLVING A CORE MELT

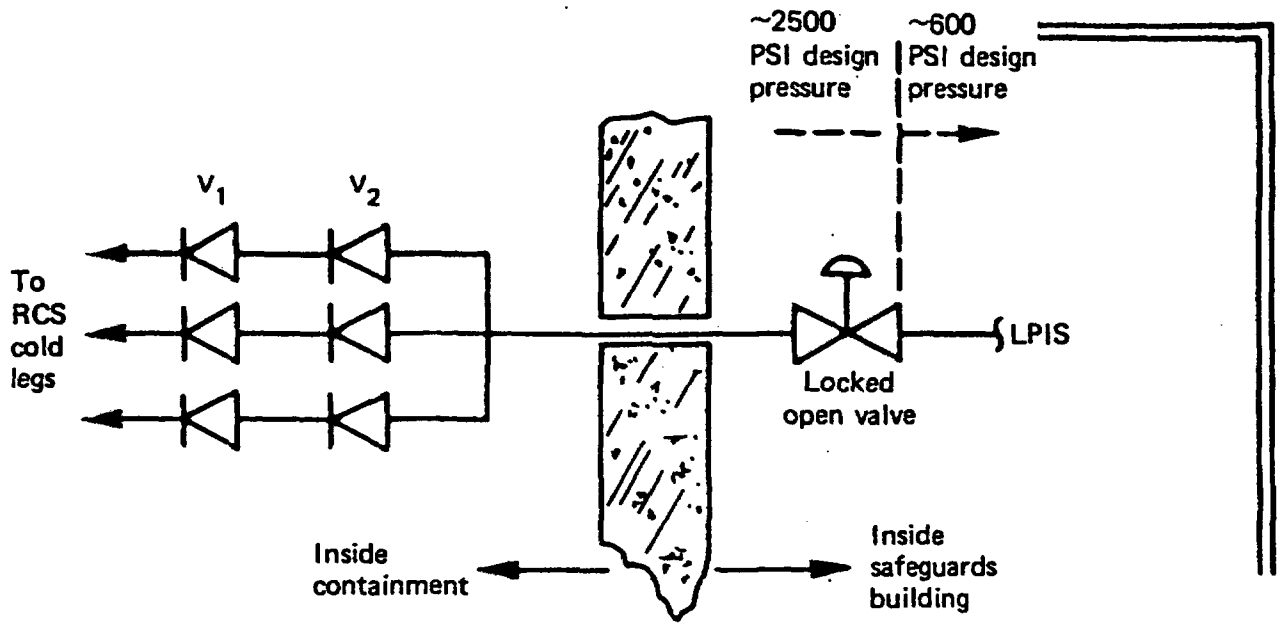

FIGURE V 4–3 PWR-LPIS CHECK VALVE DESIGN

ATTACHMENT 1

SOURCE TERM EVALUATIONS

for

POSTULATED ACCIDENT SEQUENCES

by

Battelle
Columbus Laboratories
Pacific Northwest Laboratories

Attachment 1

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Appendix V

Source Term Evaluations For Postulated Accident Sequences

by

Battelle
Columbus Laboratories
Pacific Northwest Laboratories

1. INTRODUCTION

This report presents the results of detailed calculations of fission product release from reactor containment systems for a collection of postulated accident sequences in light water-cooled power reactors. The work was done as part of and under the guidance of the AEC Reactor Safety Study.

1.1 Background

The Reactor Safety Study is composed of four major elements: (1) definition of the accident sequences and determination of their probabilities, (2) description of the physical processes and fission product source terms, (3) dispersion of the released radioactivity to the environment, and (4) assessment of the health effects and property damage. The contents of this report were developed as part of the second element which consisted of two tasks conducted by Battelle's Columbus Laboratories: (a) the degraded core analysis task, and (b) the fission product source term task. The degraded core analysis task was primarily responsible for describing the heat transfer, thermodynamic, and mechanical phenomena taking place during the various accident sequences. This information was used by the fission product source term task in models of fission product release and transport to calculate fission product escape from containment as a function of time for the accident sequences. Work on both these tasks required considerable input from the first element of the Reactor Safety Study. This included identification of potential accident sequences, specification of the status of system components at accident initiation, and selection of key sequences for detailed consequence analysis. Conversely, final results from the two tasks (containment fission product escape fractions and physical properties of the effluent) constituted direct input to the third element of the Reactor Safety Study which calculated the external distribution of the released source material. In practice, the interaction or coupling between the first three elements of the Reactor Safety Study was a continuous, iterative process in which preliminary and then refined results in each area were used in the others to develop the final analytical procedures.

1.2 Organization of Report

The following sections of this report present the final results of combined effort on the two tasks which were concerned with providing accident processes and source terms, as reported in Appendices VII and VIII of the Reactor Safety Study report. First, the scope, the analytical developments, and the input requirements of each task are summarized in order to illustrate the direct relationship between the two. Then the results of applying these techniques to the analysis of specific reactor accident sequences are given. The PWR cases are discussed first followed by the BWR cases. Every accident sequence that was calculated is identified, and both the detailed input data used and the output results obtained for each are presented.

2. ACCIDENT PROCESSES AND SOURCE TERMS

2.1 Degraded Core Analysis Task

2.1.1 SCOPE

The objective of the degraded core analysis task was to describe the physical phenomena that would be expected to occur during hypothetical reactor meltdown accidents. The principal factors and physical processes considered included:

a. The time scale of the accident, particularly the times for the initiation and completion of core melting

b. Steam generation rates during core meltdown

c. The rate and extent of zirconium-water reaction

d. The probability and consequences of hydrogen burning or exploding in the containment building

e. Probability and magnitude of steam explosions in the reactor vessel when molten core materials come into contact with water; and the likelihood that such steam explosions will be energetic enough to rupture the reactor vessel and the containment building

f. The time required for the molten core to melt through the reactor vessel

g. Probability and magnitude of steam explosions that could occur when the molten core falls to the floor of the reactor cavity; also the probability of containment failure due to these steam explosions

h. Pressure-time histories within the reactor containment building, including the times of potential containment failure due to overpressure

i. Time-dependent leak rates from the containment building

j. The interaction of the molten core with the concrete foundation and the time required to melt through the bottom of the containment building

k. The anticipated movement of the molten mass after meltthrough.

In many of the cases considered by the study, core degradation was the result of a potential large loss-of-coolant accident accompanied by the assumed failure of various combinations of engineered safety features. For the pressurized water reactor the initiating event was a double-ended severance of a cold-leg or a hot-leg pipe in the primary system. For the boiling water reactor, the severance of a recirculation line was assumed. A number of cases were also considered that involved small loss-of-coolant accidents and those transient accident sequences that were significant contributors to the core melt probability. All of these cases are included in this report.

2.1.2 GENERAL DEVELOPMENT

The degraded core analysis task was undertaken in the realization that it may not be possible to describe with certainty some of the complex physical phenomena that may take place during a meltdown accident. Where data or understanding of a particular aspect were incomplete, bounding calculations were performed to establish a physically consistent range of values for the particular variable. Typically the accident sequence was divided into discrete time intervals and the events in each interval were modeled to the extent possible. Basic assumptions of and inputs into the models were then varied to establish the bounds of uncertainty for the course and consequences of the events of interest. A complete description of the procedures used and the results obtained is contained in Appendix VIII of the Reactor Safety Study report. The paragraphs below outline the critical events and the factors that were considered in determining reactor system conditions.

2.1.2.1 Core Meltdown Phase.

The Emergency Core Cooling System (ECCS) is designed to recover the core and keep it cooled in the event of a loss-of-coolant accident. The accident sequences of interest in the present study involve sufficient failures of engineered safety features to lead to core melting. Among the principal items of interest are the times for the onset and completion of core melting. If the ECCS fails to recover and quench the core immediately after a LOCA, rapid core melting can be expected. Failure of the ECCS after the core has been quenched and recovered will involve the boiloff of the water in the reactor vessel and cause the onset of core melting, but with some delay. The decrease in decay heating with time will also affect the onset and rate of core melting. In the absence of water in the reactor vessel, the core will heat up essentially adiabatically, and the rate of heating will be controlled by the decay power level. Core melting in the presence of water will lead to the reaction of the Zircaloy cladding with steam. The energy generated by this reaction can be of the same order as decay heating and can thus have a significant influence on the rate of core heating. Further, the hydrogen generated by this reaction can have a bearing on containment failure.

2.1.2.2 Pressure Vessel Failure.

When a significant quantity of molten core material comes into contact with

water, there is some probability that a violent interaction or steam explosion can occur. Such steam explosions have the potential of rupturing the reactor vessel as well as producing projectiles with sufficient energy to penetrate the containment building. The occurrence of a reactor vessel steam explosion will result in the dispersal of a part of the core into the containment building; the remainder of the core falls to the bottom of the reactor vessel. If a steam explosion does not occur when the molten core drops into the vessel bottom head, boiloff of the water there and melt-through of the vessel will ensue.

2.1.2.3 Containment Failure.

As the core melts through the reactor vessel and falls to the floor of the reactor cavity, there is again the possibility of steam explosions. For the PWR design considered the potential of containment failure due to steam explosions in the reactor cavity was considered negligible, because of the large volume and the shielding of the containment liner by internal structures. For the BWR, this possibility had to be considered because of the much smaller containment volume. Whether or not a steam explosion takes place in the reactor cavity, the core will attack the floor and eventually melt through the concrete foundation mat.

The containment safeguards systems are designed to rapidly reduce the LOCA pressures to low levels, thus minimizing the leakage of any radioactivity that may be released to the containment atmosphere. Failure of containment safeguards in conjunction with a LOCA can result in pressures staying at elevated levels and, in many cases, containment failure by overpressurization. Containment pressure-time histories as well as times of containment failure have been evaluated for the various core meltdown accident sequences. In evaluating the potential for containment overpressurization, consideration has been given to the hydrogen generated by the zirconium-water reaction, the carbon dioxide from the decomposition of concrete, the possibility of hydrogen combustion or detonation, the pressure contributions due to the initial air content, and the pressure due to steam production.

The containment can also fail due to the lack of containment isolation, i.e., failure to close containment penetrations in the event of an accident that are normally open, or the inadvertent opening of penetrations that normally would be closed. Thus, the possible modes of reactor containment failure in the event of a core meltdown would be expected to be: lack of containment isolation, steam explosions, hydrogen combustion, overpressure, and melt-through. It is expected that melt-through would accompany all the other modes of containment failure in a meltdown accident; however, containment meltthrough would provide the principal path for radioactivity release only if the other modes of failure were avoided. The probabilities of the various modes of containment failure have been determined for each of the accident sequences of interest and are presented in a later part of this report.

2.2 Fission Product Source Term Task

2.2.1 GENERAL DEVELOPMENT

The purpose of the fission product source term task was to specify best estimate releases of fission products to the reactor containment, their distribution within containment, and their leakage from the containment barrier for each possible accident sequence. In order to accomplish this, applicable data on fission product release, transport, and deposition within reactor systems was reviewed and applied under the guidance of technical specialists in several areas of fission product behavior in containment systems. The effort resulted in development of a methodology which is described in detail in Appendix VII of the Reactor Safety Study report. Fundamentally, the method involves four core material fission product release terms for each of seven classes of fission product species, primary coolant system escape fractions for these species, and a containment processes model which generates fission product leakage rates as a function of time and accident conditions.

The four basic release terms, which apply to both PWR and BWR systems, consist of (1) the gap release component, (2) the core melt release component, (3) the vaporization release component, and (4) the core steam explosion release component. The first three components occur sequentially in the order listed for accidents involving reactor core meltdown, while occurrence of the fourth depends upon time predictions of steam explosion events. The seven classes of fission product species considered include the noble gases, the halogens (elemental and organic halide forms), the alkali metals, the tellurium group, the alkaline earths, the noble metals group, and the refractory oxide group. The release values for each of these

species are expressed as fractions of the total core inventory, respectively. Then the values must be multiplied by the fraction of the core which participates in the release process or event of interest in order to develop the proper release rate terms.

The primary coolant system escape fractions are based on consideration of three factors: (1) bulk gas flow as a driving force to sweep fission product vapors and aerosols out of the system, (2) deposition and plateout of fission products on internal structural surfaces, and (3) absorption of fission product species by ECC water. The escape fractions only apply to the release components which occur within the pressure vessel. Therefore, the gap and core melt release component values are multiplied by the escape fractions to specify the fraction of core inventory which enters the containment vessel volume.

The containment processes model provides estimates of fission product behavior in the reactor containment. It is a multi-compartment model which considers the following processes: natural transport and deposition, fission product removal by aqueous sprays, recirculating filter systems, once-through filter systems, water pool scrubbing, and leakage or exhaust from containment to the outside atmosphere. All these processes may occur simultaneously and the need for time dependent analyses required computer solution of the network of rate equations. The computer program was given the code name CORRAL, and because of containment systems differences, two versions were developed; i.e., CORRAL-PWR and CORRAL-BWR. The output from either of these programs defines the fraction of the core fission product inventory which is released to the external atmosphere as a function of time.

2.2.2 GENERAL CALCULATIONAL PROCEDURES

The calculation of the radioactive source term which escapes from the containment barrier in any accident sequence is performed with the appropriate version of the CORRAL code. The calculation begins by specifying numerous pieces of input data obtained from various sources. The number of compartments and their arrangement, volumes, surface areas, heights, etc., are determined by the type of reactor plant. The fission product release component fractions, the time for releases, and the organic iodide formation ratio are supplied according to the source term task specifications (See Appendix VII). Containment spray parameters, filter flow rates, decontamination factors for filters or the water pool and, decontamination factors associated with external leakage are obtained from typical design data or from the source term task review of these processes (See Appendix VII). The times for all events in the accident sequence along with containment physical conditions as a function of time are provided by the degraded core analysis task (See Appendix VIII). The physical conditions include pressure, temperature, and composition of the internal atmosphere, flow rates between compartments, leak rates to the external atmosphere, and the sizes of puff-type releases from containment.

After the input is complete and initial conditions have been set, the code uses the data to continuously compute changing properties and fission product removal rates as a function of time. These values are used in incremental solutions to the coupled set of differential equations to obtain the time dependent fission product concentrations and accumulations in each compartment of the containment.

The output from CORRAL calculations consists principally of cumulative fractional releases from containment versus time for each of the fission product groups. Internal airborne concentrations and dose reduction factors can also be supplied if desired. The releases are identified as occurring at ground level or at stack height (where appropriate), and the temperature of the released gases is provided to assist subsequent estimations of plume buoyancy. Actual accident time values are also specified so that radioactive decay corrections can be made by the next element in the analysis chain. At this point the work of the source term task is complete. The sections which follow present a detailed description of the accident sequence calculations that have been performed with the CORRAL code for both PWR and BWR systems.

3. PWR ACCIDENT SEQUENCE CALCULATION RESULTS

3.1 PWR Accident Event Trees

Power reactors of current design incorporate many safety systems which are designed to control or mitigate the consequences of the various potential accidents. The Design Basis Accident (DBA) is intended to provide an upper limit to the potential consequences of any accidents that can be considered to have a plausible chance of occurrence.

For the consequences of any accident to exceed those of the DBA, the initiating event must be more severe than any considered in the design evaluation, or it must be accompanied by the failure of at least some of the engineered safety features. The latter type of accidents constitutes the principal interest in this report. The exact course and consequences of any of this group of accidents will depend on the specific safety features which are assumed to fail in conjunction with the initiating event. Because of the multiplicity of safety features provided, many different accident sequences can be postulated. In order to provide a convenient method of delineating specific combinations of initiating events and safeguards operability, the Reactor Safety Study has utilized decision tree methodology. This has led to the development of event trees for the various initiating accidents of interest. The event tree is a diagram which shows various possible accident routes depending on which safety systems, singly or in combination, are assumed to be available when required.

The basic PWR large LOCA event tree developed by the Reactor Safety Study is illustrated on the left side of Fig. 1, the notation is defined in Table 1. The rationale behind the development of this event tree is described in Appendix I of the Reactor Safety Study report. It can be seen that this event tree identifies 45 specific combinations of safety system failures or successes occurring in conjunction with the large LOCA. Upward branches in the tree represent success and downward branches represent failure. Each of these combinations, called sequences, can lead to differences in accident conditions and eventual consequences, and each will of course have a different probability of occurrence. Core melting is predicted for 41 of these sequences. The calculations discussed in this report deal primarily with the core melt sequences.

If a core meltdown does occur, the course of the accident can be further divided into several alternate paths, each representing a particular mode of containment failure. The identification of an accident sequence is not complete until the particular containment failure mode is specified. These possible failure modes are illustrated in Fig. 2 and also indicated across the top of the right side of Fig. 1. The particular failure modes that are applicable in each accident sequence are identified by an "x" in the appropriate column in Fig. 1. The rationale behind the assignment of these various combinations is given in Appendix VIII of the Reactor Safety Study report. It may be noted that for accident sequences involving core meltdown, containment failure by one or a combination of the possible modes must necessarily follow.

Event trees for small break LOCA's and transients have also been developed; their description and rationale behind their development are given in Appendix I. The notation used in the small break and transient event trees is defined in the body of Appendix V. The potential containment failure modes for these sequences are the same as for the large LOCA.

Table 2 summarizes the results of the PWR degraded accident analyses. The accident sequence and containment failure mode designations are the same as previously described. The results show the estimated time periods for core meltdown, pressure vessel meltthrough, and containment meltthrough along with the containment internal pressures at these times. In sequences where containment failure by overpressure can occur, the estimated failure time is listed. The probability of failure by overpressure and the probabilities of the other modes of containment failure are also given. Some of these failure modes are assumed to occur at particular time points in the sequences. Thus, by definition, failures due to steam explosions in the pressure vessel happen only at the end of core meltdown, failures by lack of isolation are coincident with the blowdown time that immediately follows the large pipe break, and failures by containment meltthrough occur only at the end of the meltthrough period. However, failures by overpressure, including the effects of hydrogen combustion, do not occur at a predetermined point in time, because this mode depends on the particular containment conditions in each sequence.

Evaluation of the data summarized in Table 2 indicated similarities in conditions among many of the potential 155 core melt sequences. Therefore, it was decided to perform detailed fission product release calculations for only a selected but representative number of sequences. The logic involved in selecting the key sequences for detailed calculation is described in the body of Appendix V. The results of this selection for large LOCA's are indicated in Fig. 1 by the circled x's. Altogether, 38 sequence cases were selected for detailed calculation using the CORRAL-PWR computer code.

3.2 Input Data Used in the CORRAL-PWR Calculations

In order to perform the calculations of fission product escape from containment as a function of time with the CORRAL code, a number of different types of input data are needed. These include times and magnitudes for important events and processes that occur after the initial pipe break. In this section the key input data used for each of the sequences that were calculated are given.

The first set of data concerns critical information on the time schedule of key events in each accident sequence. Most of this information appears in Table 2. Additional data regarding containment conditions are required, including leak rates to the outside atmosphere as a function of time, and specification of containment spray operation. These are given in Table 3. Note that the accident histories are divided into various time intervals and the occurrence of rapid containment depressurizations (puff discharges) are specifically identitied along with the percentage of the contained volume that escapes at each puff discharge.

The third set of required data defines the fission product release fractions that occur during the course of the accident sequences. The release terms used in the 38 sequence calculations are given in Table 4. Each value specifies the fraction of the core fission product inventory for the element which is released to the containment vessel atmosphere over the indicated time period. The gap and steam explosion components are treated as instantaneous releases, while the melt release is assumed to occur at a constant rate over the core meltdown time period. The vaporization release is assumed to follow a decreasing exponential rate over the first two hours of the containment base mat melt-through period (See Appendix VII). In addition, all fission product release fractions for steam explosions assume that only one-half the fuel, which is predicted to be molten at the time, actually participates in the event such that small pieces are ejected into the containment atmosphere and oxidized. It should be noted that some slight differences are evident among release fractions for equivalent components of different sequence cases. These differences arose because most of the CORRAL calculations were done before the fission product release methodology was finalized. However, since the differences cause only minor perturbations in CORRAL results, it was considered unnecessary to re-do calculations using the finalized input.

The last set of input data needed for the CORRAL calculations consists of a collection of constants and parameters which define the physical properties of the PWR containment system. These include such items as compartment volumes, surface areas, heights, and containment spray operation parameters. In addition, fission product behavior factors such as decontamination factors during leakage and the organic iodide conversion ratio are supplied. Table 5 lists the input values used in the CORRAL calculations for these various parameters.

3.3 Results of the CORRAL-PWR Calculations

The output from CORRAL calculations consists of fractions of core inventory released from containment to the outside atmosphere as a function of time after the accident initiating event. Release fractions are calculated for eight different groups of fission products; the noble gases (Kr-Xe), organic iodide (Org-I), elemental halogens (I-Br), alkali metals (Cs-Rb), the tellurium group (Te), alkaline earths (Ba-Sr), the noble metal group (Ru), and the refractory oxides (La). The results obtained for each of the 33 sequences calculated appear in Table 6 where cumulative core fractional releases are given at important time points throughout the postulated accident histories. These data were supplied directly to the next element of the Reactor Safety Study - the atmospheric dispersion analysis effort.

The intepretation of the significance of all these results must be made within the context of the total Reactor Safety Study program. Detailed analysis of the data in Table 6 requires careful attention to the accident scenario and the physical conditions for each sequence. Scenarios are briefly outlined in Appendix I of the Reactor Safety Study report, and a summary of physical conditions are given in Tables 2 and 3. However, a general examination of the data in Table 6 indicates several useful observations about the effect of internal processes within the reactor containment system. For example, results for the noble gases usually indicate almost total release from the containment system. This is because containment processes are assumed not to exert any removal of these species from the internal atmosphere. Their escape, as a function of time, depends only on the relation-

ship between their rate of initial release from the fuel and the rate of volumetric leakage from the containment structure. Therefore, in cases of containment meltthrough from an isolated containment (ε paths), less than total release occurs in 30 days only because the leakage has not been sufficient to exhaust all the airborne contents from the containment vessel. The organic iodide release behaves in an identical manner, except its ultimate core fractional release is limited by the conversion ratio (organic iodide:total iodine) that is used in the calculation. All the other fission product species experience removal from the internal containment atmosphere by one or more mechanisms. Some of these components also are expected to be partially retained by the reactor fuel (see Table 4). Therefore, the core fractional releases of these components must be less than the corresponding noble gas values. The size of the difference in each case, taking into account differences in the initial releases from the fuel, is indicative of the effectiveness of the removal processes in reducing the external source term for each species.

It is convenient to review the results for these "removable species" by grouping cases which consider similar events such as the modes of containment failure.

a. Steam Explosion Cases. A total of nine steam explosion sequences were calculated. In five of these, (i.e., AHα, AHIα, ADα, ADIα, and ACDGIα) containment sprays (CSIS and/or CSRS) were operating during the period of major fission product release. In the other four (ABα, AHFα, $S_2C\alpha$, and TMLB'α), containment sprays were not operating. Inspection of the results shows that the primary advantage of the sprays in limiting fission product releases to the atmosphere at the steam explosion, comes from the containment pressure reduction they produce which then causes a smaller postulated pressure puff at containment failure. The fission product scavenging effect of the sprays is generally of secondary importance. The presence of sodium hydroxide in the spray solution reduces iodine-bromine releases only, and in these cases by a factor of about three.

b. Overpressure Cases. Nine containment overpressure failure sequences were calculated, but in one of these (AGδ) the failure precedes any major fission product releases from the reactor fuel. The other eight sequences consist of two in which containment sprays operate (i.e., AHGδ and ACDGIδ), and six in which sprays do not operate (i.e., ABδ, ABγ, AHFδ, $S_2C\delta$, TMLB'δ, and TMLB'γ). Comparison of the core fractional releases to the atmosphere for these two conditions shows that the sprays are very effective in reducing the atmospheric releases. This is because the time required to reach overpressure conditions provides ample opportunity for the sprays to reduce the airborne concentrations of the fission products in the containment vessel.

c. Lack of Isolation Cases. Seven of these cases were calculated, including five in which containment sprays operate (i.e., AHβ, AHIβ, ACDβ, ADβ, and ACDGIβ), one in which sprays operate part of the time (ADFβ), and one in which no sprays operate during major fission product release periods (AHFβ). In all cases, core meltdown, pressure vessel meltthrough, and containment meltthrough proceed sequentially in a containment vessel which has a large leak rate. Comparison of results for the different sequences indicates the containment sprays effectively compete with the large leak rate to limit fission product losses to the external atmosphere. On the other hand, if the sprays do not operate (AHFβ), the results show that natural deposition processes within the containment system are not very effective in reducing the magnitude of the external atmospheric source term.

d. Containment Meltthrough Cases. A total of twelve meltthrough sequences were calculated including nine in which containment sprays operate (i.e., AHε, AHIε, ADε, ADIε, AHGε, ADGε, ADGIε, ACDε, and ACDGIε), one in which sprays operate only part of the time (ADFε), and two in which sprays do not operate during major fission product release periods (i.e., AHFε and ABε). The fission product atmospheric release values for all the spray operation sequences are very similar, and the results indicate that spray operation is quite effective in reducing releases of the removable species. The atmospheric release values for the "partial spray" and "no spray" operation sequences are also very similar, but significantly higher than for the "spray operation" cases. These release values reflect

the effects of two decontamination processes: natural deposition within the containment structure before meltthrough, and the decontamination which is associated with transport of airborne contents through the ground to the atmosphere when meltthrough occurs. (It is assumed that meltthrough is immediately followed by rapid relief of the containment pressure through the ground.) Although a DF value of 1000 was used for this latter process, analysis of the calculated results indicates the overall effect on the cumulative releases ranging from a factor of only 5 to about 50. This is because releases to the atmosphere due to nominal leakage before meltthrough make a significant contribution to the total releases. Natural deposition processes within the containment before meltthrough can also produce rather significant reductions in the internal airborne fission product concentrations.

One final general observation regarding the data in Table 6 is worth noting. The cumulative fission product release to the atmosphere changes very little between about one day after the accident starts and thirty days. The only exceptions to this rule are the noble gas and organic iodide values for some sequences in which long-term leakage from the containment can eventually exhaust these nonremovable forms. This observation, therefore, indicates that nearly all of the immediate effects of an accident will be experienced within a day from the start of the event.

4. BWR ACCIDENT SEQUENCE CALCULATION RESULTS

4.1 BWR Accident Event Trees

The BWR large LOCA event tree developed by the Reactor Safety Study is shown on the left side of Fig. 3; the notation is explained in Table 7. This event tree identifies 30 separate combinations of safety system operation or failure occurring in conjunction with a large pipe break. Each of these combinations, or accident sequences, can have different conditions and consequences. Therefore, the probability of each sequence can also be expected to be different. Core melting is predicted for 24 of the sequences.

As in the PWR accident sequences previously discussed, the occurrence of core meltdown in a BWR can lead to several alternate paths to containment failure. The events considered in the BWR core melt event tree, shown in Fig. 4, include primary containment isolation failure in the drywell, primary containment isolation failure in the wetwell, reactor vessel steam explosions, containment steam explosions, primary containment failure due to overpressure, secondary containment failure, and standby gas treatment system filter failure. These events are considered singly and in various combinations. The BWR core melt event tree differs from the PWR containment failure modes tree largely because of differences in the containment design. The BWR pressure suppression containment consists of two distinct parts and the location of potential failures can affect accident consequences; thus the two locations of containment isolation failure must be considered separately. Steam explosions in the reactor cavity were found to have a negligible potential for containment failure in the PWR; but for the BWR the probability of this failure mode, though small, should be considered. Hydrogen combustion has the potential to contribute to containment failure in the PWR, but not in the BWR. This is a consequence of the inerted atmosphere in the BWR containment. Secondary containment with its standby gas treatment system can mitigate the consequences of BWR accidents. Thus the operability and/or failure of these must be included in the evaluation. Containment meltthrough is expected to accompany all the core meltdown accidents in the BWR, as was the case with the PWR. However, it has been found that meltthrough would not be a primary mode of containment failure in any of the BWR accident sequences; consequently, it does not appear in the core melt event tree.

The various possible containment failure modes are indicated in the headings across the top of the right side of Fig. 3. The particular failure modes that are applicable to each of the accident sequences are identified by an "x" in the appropriate column under these headings. The basis for the determination of the various combinations is given in Appendix VIII of the Reactor Safety Study Report.

Accident event trees have also been developed for BWR small break LOCA's and transients; the rationale for their development is given in Appendix I. The notation used for the small LOCA and transient events is also given in the body of this Appendix. The potential containment failure modes are the same as for the large LOCA.

The timing of the principal events for the various BWR core melt accident sequences is given in Table 8. These include the times for core meltdown, reactor vessel meltthrough, and containment meltthrough along with the containment pressures at these times. In sequences where containment failure by overpressure can occur, the time of this failure is given. Except for sequences involving failure of emergency core cooling function and failure to scram, reactor vessel steam explosions, if they occur, are assumed to take place at the end of core meltdown. In the former cases, steam explosions are assumed to occur when the hotter half of the core is molten, or about 20 minutes into the accident. Containment steam explosions would be expected at the time of reactor vessel meltthrough. Containment isolation failures are assumed to have taken place prior to the LOCA. The probabilities of the various BWR core melt sequences are given in Table 9.

Evaluation of the data summarized in Table 8 indicated similarities in conditions among many of the potential 116 sequences. Therefore, detailed calculations of fission product release were performed for only a selected, but representative, number of sequences. The logic involved in selecting the key sequences for detailed calculation is described in the body of Appendix V. The results of this selection are indicated in Fig. 3 by the circled x's. Altogether, 23 sequence cases were picked for detailed calculation using the CORRAL-BWR computer code, and 6 additional cases were selected for limited calculation using simplified hand computations. In addition, one sequence with no core melting was calculated with CORRAL: sequence A, in which only the gap release occurs and all safety systems operate after the pipe break.

4.2 Input Data Used in the CORRAL-BWR Calculations

In order to perform the calculations of fission product escape from containment as a function of time with the CORRAL code, a number of different types of input data are needed. These include times and magnitudes for important events and processes which occur after the initial pipe break. In this section the key input data used for each of the sequences that were calculated are given.

The first set of required data defines the time schedule of key events in each accident sequence. Most of this information appears in Table 8. Additional data regarding containment conditions are required, including leak rates from primary containment as a function of time, and volumetric flows from the drywell to the suppression pool. These data are given in Table 10. Note that the accident histories are divided into various time intervals, and the occurrence of rapid containment depressurizations (puff discharges) are specifically identified along with the percentage of the contained volume that escapes at each puff discharge.

The next set of required data defines the fuel fission product release fractions that occur during the accident sequences. The release terms used in all the sequence calculations are given in Table 11. The gap and steam explosion components are treated as instantaneous releases, while the melt release is assumed to occur at a constant rate over the core meltdown time period. The vaporization release is assumed to follow a decreasing exponential rate over the first two hours of the containment meltthrough period.[1] Several factors had to be considered in developing release terms for the BWR calculations. For example, core meltdown is expected to occur in two stages with different rates in sequences involving emergency core cooling function (ECF) failure (see Table 10). Since the CORRAL code is not programed to accept two melt release rates, the release for the initial period of rapid melting was combined with the gap release and input to the computer calculation as "gap release". Then the release associated with the subsequent period of slower core melting was input as the "melt release". Furthermore, the values for both of these release components include the anticipated effects of fission product scrubbing by the ECC water.[1] For sequences in which emergency core cooling fails to operate (ECO) and dry core meltdown follows, only two-thirds of the fission product melt release is assumed to escape the pressure vessel during the meltdown period.[1] However, the remaining one-third is always added to the specified releases for the next release component in the accident sequence; i.e., either the containment steam explosion release or the vaporization release. This is because the conditions which accompany both these releases should effectively sweep the residual fission products from the pressure vessel. All

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[1]See Appendix VII.

fission product release values for the steam explosion component, when the steam explosion occurs at the bottom of the drywell as opposed to in the pressure vessel, take into account the limited oxygen that would be available in an inerted BWR containment to cause oxidation of the UO_2 fuel. Since there is sufficient oxygen to oxidize only about 7 percent of the total fuel, the fission product release fractions reflect this limit. Finally, all fission product releases at steam explosions assume that only one-half the fuel, which is predicted to be molten at the time, actually participates in the event. It should be noted that some slight differences are evident among release fractions for equivalent components of different sequence cases. These differences arose because most of the CORRAL calculations were done before the fission product release methodology was finalized. However, since the differences cause only minor perturbations in CORRAL results, it was considered unnecessary to re-do calculations using the finalized inputs.

The last set of input data needed for the CORRAL calculations consists of a collection of constants and parameters which define the physical properties of the BWR containment system. These include such items as compartment volumes, surface areas, and heights, and the standby gas treatment system performance data. In addition, fission product behavior data such as decontamination factors, particulate size specifications, and the organic iodide conversion ratio are supplied. Table 12 lists the input values used in the CORRAL calculations.

4.3 Results of the CORRAL-BWR Calculations

As noted in the section on PWR calculations, the output from CORRAL consists of fractions of core inventory released from containment to the outside atmosphere as a function of time after the accident initiating event. Release fractions are calculated for eight different fission product groups: the noble gases (Kr-Xe), organic iodides (Org-I), elemental halogens (I-Br), alkali metals (Cs-Rb), the tellurium group (Te), alkaline earths, (Ba-Sr), the noble metals group (Ru), and the refractory oxides (La). The results obtained for each of the 15 sequences calculated with CORRAL-BWR appear in Table 13, where cumulative core fractional releases are given at important time points throughout the postulated accident histories. In addition, abbreviated results from the simplified hand calculation of 6 other sequences are tabulated in Table 14. These hand calculations used the same basic geometry as CORRAL for the BWR containment system, but the network of CORRAL rate equations were decoupled and solved individually to provide an approximate method for following fission product transport through the several compartments to the atmosphere. The fission product removal coefficients and decontamination factors used in the hand calculations were estimated from analysis of CORRAL data for similar accident situations. Experience indicates that hand-calculated results, of the type given in Table 14, agree with CORRAL results to within a factor of 2 or 3. This was considered sufficient for the sequences that were calculated, because their probability of occurrence was expected to be quite low.

Interpretation of the significance of all the fission product release results must be made within the context of the total Reactor Safety Study program. The estimated impact of such releases and the estimated probability of realizing the various accident sequences need to be included. Furthermore, any detailed analysis of the data from Tables 13 and 14 requires careful attention to the individual accident scenarios and conditions. Scenarios for the BWR are briefly outlined in Appendix I, and a summary of physical conditions is given in Tables 8 and 10. Nevertheless, a general examination of the fission product release results indicates several useful observations about key events or processes which deserve comment.

First, the examination readily reveals that most of the release values are greater than 1 percent of the core inventory. The exceptions always include the refractory oxides (La), almost always include the alkaline-earths (Ba-Sr), and frequently include the noble metals (Ru). These exceptions are primarily a result of the lower initial releases from the reactor fuel for these species compared to the other species. Hence, it may be concluded that internal decontamination processes in a BWR have about the same effect on all fission products which can be removed from the gas phase. If violent failure of the primary containment, resulting in direct leakage to the atmosphere, occurs, then the internal decontamination processes are not very helpful in reducing the external source term. This is illustrated by the results for cases involving

steam explosions (i.e., AEα (wet), AEβ (dry), AFα, and AFβ) and by results for cases involving overpressurization (i.e., AEγ (dry) and AFγ, and also AFγ, ADJγ, ADFγ, and ADEγ (dry) when these latter cases lead to rupture of an outside wall in the lower secondary containment structure). However, if the leakage path of primary containment gases is through the suppression pool, or through the lower parts of secondary containment (the annulus), and/or from an intact reactor building, the internal decontamination processes can reduce the atmospheric source term by factors ranging from about 10 to 100. This is illustrated by results for most of the isolation failure cases (i.e., AGJδ, AGJδθ, AEGδ (dry), AEGδη (dry), AGJεζ, AGJεη, AFGεζ, and AFGεη) and the results for overpressure failure cases, if the overpressurization occurs before core melting and the outside walls of the lower secondary containment remain intact (i.e., AJγ, ADJγ, ADFγ, and ADEγ (dry)).

The principal internal decontamination processes in these BWR analyses consisted of natural deposition in the primary containment volume, fission product absorption (scrubbing) in the pressure suppression pool, and natural deposition in the annular air space. For elemental iodine, deposition in the annulus usually constituted the major decontamination process, while for the particulate fission products, deposition in the primary containment and in the annulus were about equally effective. Fission product absorption in the suppression pool was usually unimportant except for sequences which involved primary containment isolation failure. In three of the calculated sequences (i.e., A, AGJε, and AFGε), the standby gas treatment system filters were effective in reducing the magnitude of the atmospheric releases. This was because all external leakage from the reactor system passed through the filters, and the predicted fission product loadings did not cause the filter overheating criteria to be exceeded. In two calculated sequences (i.e., AGJδ and AEGδ (dry)) the filters were only partially effective because ground-level leakage from the reactor building was predicted to occur. Ground-level leakage takes place in the analysis only when gas flow rates into the secondary containment are greater than the exhaust capacity of the SGTS fans (10,000 cfm). In addition, the charcoal beds (but not the HEPA filters) were predicted to exceed the overheating criteria at about 7 hr. in each case. This condition in CORRAL calculations is simply assumed to render the charcoal beds incapable of any additional iodine removal as explained in Appendix VII.

TABLE 1 PWR EVENT TREE SYMBOLS

Letter	Symbol	Meaning
A	LPB	Large Pipe Break
B	EP	Electric Power
C	CSIS	Containment Spray Injection System
D	ECI	Emergency Core Cooling Injection
E	ECF	Emergency Core Cooling Function
F	CSRS	Containment Spray Recirculation System
G	CHRS	Containment Heat Removal System
H	ECR	Emergency Core Cooling Recirculation
I	SHA	Sodium Hydroxide Addition
α	CRVSE	Containment Rupture due to a Reactor Vessel Steam Explosion
β	CL	Containment Leakage
γ	CR-B	Containment Rupture due to Hydrogen Burning
δ	CR-OP	Containment Rupture by Overpressurization
ε	CR-MT	Containment Rupture by Meltthrough

TABLE 2 SUMMARY OF PWR DEGRADED ACCIDENT ANALYSES

Sequence	Start Boiloff, min	Core Melting Time, min Start	Core Melting Time, min End[a]	Core Melting Pressure, psia w/o H_2 Comb.	Core Melting Pressure, psia w/H_2 Comb.[b]	Reactor Vessel Meltthrough Time, min	Reactor Vessel Meltthrough Pressure, psia	Containment Overpressure Failure, min	Containment Meltthrough Start, min[c]	Containment Meltthrough Pressure, psia	Containment Meltthrough End, min	Containment Meltthrough Pressure, psia	Containment Failure Mode Probabilities[d] α CRVSE	β CL	γ CR-B[e]	δ CR-OP	ε CR-MT
A	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
AI	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
AR	60	100	150	16	13	210	16	-	230	18	1290	29	0.01	0.00	0	0	0.99
ARI	60	100	150	16	13	210	16	-	230	18	1290	29	0.01	0.00	0	0	0.99
AG	1290	1370	1490	15	15	1610	15	1290	1630	15	3800	15	0.01	0.00	0	1.00	0
AGI	1290	1370	1490	15	15	1610	15	1290	1630	15	3800	15	0.01	0.00	0	1.00	0
ARG	60	100	150	16	17	210	16	1280	230	18	1290	100	0.01	0.00	0	0.49	0.49
ARGI	60	100	150	16	17	210	16	1280	230	18	1290	100	0.01	0.00	0	0.49	0.49
AF	530	590	670	15	15	760	15	530	790	15	2400	15	0.01	0.00	0	1.00	0
AFI	530	590	670	15	15	760	15	530	790	15	2400	15	0.01	0.00	0	1.00	0
ARF	60	100	150	46	65	210	40	230	230	80	1290	45	0.01	0.00	0.12	0.08	0.78
ARFI	60	100	150	46	65	210	40	230	230	80	1290	45	0.01	0.00	0.12	0.08	0.78
AE	1	16	60	14	11	120	16	-	140	18	1200	29	0.01	0.00	0	0	0.99
AEI	1	16	60	14	11	120	16	-	140	18	1200	29	0.01	0.00	0	0	0.99
AEG	1	16	60	15	22	120	16	1280	140	18	1200	94	0.01	0.00	0	0.39	0.60
AEGI	1	16	60	15	22	120	16	1280	140	18	1200	94	0.01	0.00	0	0.39	0.60
AEF	1	16	60	16	24	120	16	-	140	18	1200	50	0.01	0.00	0	0	0.99
AEFI	1	16	60	16	24	120	16	-	140	18	1200	50	0.01	0.00	0	0	0.99
AD	1	16	60	14	11	120	16	-	140	18	1200	29	0.01	0.00	0	0	0.99
ADI	1	16	60	14	11	120	16	-	140	18	1200	29	0.01	0.00	0	0	0.99
ADG	1	16	60	15	22	120	16	1230	140	18	1200	94	0.01	0.00	0	0.39	0.60
ADGI	1	16	60	15	22	120	16	1230	140	18	1200	94	0.01	0.00	0	0.39	0.60
ADF	1	16	60	16	24	120	16	-	140	18	1200	50	0.01	0.00	0	0	0.99
ADFI	1	16	60	16	24	120	16	-	140	18	1200	50	0.01	0.00	0	0	0.99
AC	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
ACI	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
ACH	120	170	220	16	13	280	16	-	300	18	1360	25	0.01	0.00	0	0	0.99
ACHI	120	170	220	16	13	280	16	-	300	18	1360	25	0.01	0.00	0	0	0.99

TABLE 2 (Continued)

Sequence	Start Boiloff, min	Core Melting Time, min Start	Core Melting Time, min End[a]	Core Melting Pressure, psia w/o H_2 Comb.	Core Melting Pressure, psia w/H_2 Comb.[b]	Reactor Vessel Meltthrough Time, min	Reactor Vessel Meltthrough Pressure, psia	Containment Overpressure Failure, min	Containment Meltthrough Start, min[c]	Containment Meltthrough Pressure, psia	Containment Meltthrough End, min	Containment Meltthrough Pressure, psia	Containment Failure Mode Probabilities[d] α CRVSE	β CL	γ CR-B[e]	δ CR-OP	ε CR-MT
ACG	1290	1370	1490	15	15	1610	15	1290	1630	15	3800	15	0.01	0.00	0	1.00	0
ACGI	1290	1370	1490	15	15	1610	15	1290	1630	15	3800	15	0.01	0.00	0	1.00	0
ACHG	120	170	220	16	17	280	22	1280	300	24	1360	107	0.01	0.00	0	0.60	0.39
ACHGI	120	170	220	16	17	280	22	1280	300	24	1360	107	0.01	0.00	0	0.60	0.39
ACF	240	290	360	15	15	440	15	240	460	15	1900	15	0.01	0.00	0	1.00	0
ACHF	120	170	220	105	130	280	95	200	300	110	1360	75	0.01	0.00	0.24	0.56	0.19
ACE	1	16	60	43	55	120	16	-	140	18	1200	29	0.01	0.00	0	0	0.99
ACEI	1	16	60	43	55	120	16	-	140	18	1200	29	0.01	0.00	0	0	0.99
ACEG	1	16	60	43	55	120	16	1280	140	22	1200	94	0.01	0.00	0	0.39	0.60
ACEGI	1	16	60	43	55	120	16	1280	140	22	1200	94	0.01	0.00	0	0.39	0.60
ACEF	1	16	60	75	100	120	65	60	140	75	1200	60	0.01	0.00	0.12	0.04	0.82
ACD	1	16	60	43	55	120	16	-	140	18	1200	29	0.01	0.00	0	0	0.99
ACDI	1	16	60	43	55	120	16	-	140	18	1200	29	0.01	0.00	0	0	0.99
ACDG	1	16	60	43	55	120	16	1200	140	24	1200	100	0.01	0.00	0	0.49	0.49
ACDGI	1	16	60	43	55	120	16	1200	140	24	1200	100	0.01	0.00	0	0.49	0.49
ACDF	1	16	60	75	100	120	65	60	140	75	1200	60	0.01	0.00	0.12	0.04	0.82
AB	1	16	60	75	100	120	65	60	140	75	1200	60	0.01	0.00	0.12	0.04	0.82
S_2C	240	290	360	15	15	440	15	240	460	15	1900	15	0.01	0.00	0	1.00	0
TMLB'	120	170	220	105	130	280	95	200	300	110	1360	75	0.01	0.00	0.24	0.56	0.19

(a) End of core melting is taken as ~80 percent of the core molten.
(b) Assuming the hydrogen from the reaction of 75 percent of the cladding with steam burns as it is generated.
(c) After the initial rapid interaction of the molten core with concrete.
(d) The probabilities apply to the primary modes of containment failure; meltthrough is expected to occur in all cases.
(e) 0.00 indicates that there exists no significant figure until the third decimal place. Probability values for containment leakage were derived from fault tree analyses presented in Appendix II and the median values were 2×10^{-3} or less.

TABLE 3 PWR TIME DEPENDENT LEAK RATES

Subsequence	CSIS Start, min.	CSIS End, min.	CSRS Start, min.	CSRS End, min.	Puff Release Time, min.	Puff Release Fraction, v/o[a]	Leakage Time Interval, min.	Leakage Leak Rate, v/o/hr[a]	Leakage Pressure, psia	Leakage Temperature F	Remarks
AHα	0.5	47	3.3	--	150	8.1	0-10	4.2×10^{-2}	25	205	Nominal leakage, gap release
							-100	4.2×10^{-3}	15	150	Pressure decreased by sprays
							-150	4.2×10^{-3}	16	150	Core melting, pressure due to H_2
							-210	0	15	150	Reactor vessel melting
							-230	8	15	150	Leakage of Co_2 from concrete
							-420	1.4	15	150	Leakage of CO_2 from concrete
							-900	1.1	15	150	Leakage of CO_2 from concrete
							-1800	0.83	15	150	Leakage of CO_2 from concrete
AHε	0.5	47	3.3	--	--	--	0-10	4.2×10^{-2}	25	205	Nominal leakage, gap release
							-100	4.2×10^{-3}	15	150	Pressure decreased by sprays
							-150	4.2×10^{-3}	16	150	Core melting, pressure due to H_2
							-210	4.2×10^{-3}	16	150	Reactor vessel melting
							-500	8.4×10^{-3}	17	150	Leakage of CO_2 from concrete
							-800	1.7×10^{-2}	19	150	Leakage of CO_2 from concrete
							-1290	2.5×10^{-2}	29	150	Leakage of CO_2 from concrete
							1290-	4.2×10^{-2}	34	150	Pressure set by external head
AHβ	0.5	47	3.3	--	--	--	0-10	42	25	205	Large leak, gap release
							-60	4.2	15	150	Pressure decreased by sprays and leakage
							-100	4.2	15	150	Leakage of noncondensables
							-150	18	15	150	Core melting, leakage of noncondensables
							-21	0	15	150	Reactor vessel melting
							-230	16	15	150	Leakage of CO_2 from concrete
							-420	2.7	15	150	Leakage of CO_2 from concrete
							-900	2.0	15	150	Leakage of CO_2 from concrete
							-1800	1.7	15	150	Leakage of CO_2 from concrete
AGδ	0.5	47	3.3	1290	1290	85	0-10	4.2×10^{-2}	42	245	Nominal leakage, gap release
							-30	2.1×10^{-2}	14	145	Pressure decreased by sprays
							-130	0		150	No driving force for leakage
							-240	2.1×10^{-2}	25	205	Heating of water
							-1290	4.2×10^{-2}	100	315	Heating of water, containment failure
							-1370	28	15	212	Boiloff of water in reactor vessel
							1370-1490	28	15	212	Core melting, continued boiloff
							-1610	0	15	212	Reactor vessel melting
							1610-	41	15	212	CO_2 and H_2O from concrete

TABLE 3 (CONTINUED)

Subsequence	CSIS Start, min.	CSIS End, min.	CSRS Start, min.	CSRS End, min.	Puff Release Time, min.	Puff Release Fraction, v/o[a]	Leakage Time Interval, min.	Leakage Leak Rate[b] v/o/hr	Leakage Pressure, psia	Leakage Temperature F	Remarks
ABα	--	--	--	--	60	80	0-60	4.2×10^{-2}	75	291	Core melting, nominal leakage
							-120	0	15	212	Reactor vessel melting
							-140	25	15	212	CO_2 and H_2O from concrete
							-360	4.8	15	212	CO_2 and H_2O from concrete
							-420	4.1	15	212	CO_2 and H_2O from concrete
							-900	3.0	15	212	CO_2 and H_2O from concrete
							-1800	2.6	15	212	CO_2 and H_2O from concrete
ABγ	--	--	--	--	60	85	0-60	4.2×10^{-2}	100	315	Nominal leakage until containment fails
							-75	320	15	212	Boiloff of water in reactor vessel
							-120	0	15	212	Reactor vessel melting
							-140	50	15	212	CO_2 and H_2O from concrete
							-420	9	15	212	CO_2 and H_2O from concrete
							-900	6	15	212	CO_2 and H_2O from concrete
							900-	5	15	212	CO_2 and H_2O from concrete
ABε	--	--	--	--	1200	45	0-60	4.2×10^{-2}	75	291	Core melting, nominal leakage
							-120	4.2×10^{-2}	65	280	Reactor vessel melting
							-140	4.2×10^{-2}	75	291	Concrete decomposition
							-1200	4.2×10^{-2}	60	271	Concrete decomposition
							1200-	4.2×10^{-2}	34	218	Pressure set by external head
ABδ	--	--	--	--	--	--	0-16	42	60	271	Core heating, large leak
							-60	42	50	259	Core melting, large leak
							-120	42	25	240	Reactorvessel melting
							-140	42	25	240	Initial concrete attack
							-360	9.7	15	212	CO_2 and H_2O from concrete
							-420	8.2	15	212	CO_2 and H_2O from concrete
							-900	6.0	15	212	CO_2 and H_2O from concrete
							-1800	5.2	15	212	CO_2 and H_2O from concrete
ACDIα	--	--	3.3	--	60	65	0-16	4.2×10^{-2}	56	268	Core heating, nominal leakage
							-60	4.2×10^{-2}	43	246	Core melting, nominal leakage
							-120	0	15	212	Reactor vessel melting
							-140	119	15	212	Boiloff and concrete decomposition
							140-210	99	15	212	Boiloff and concrete decomposition
							210-	74	15	212	Boiloff and concrete decomposition

TABLE 3 (CONTINUED)

Subsequence	CSIS Start, min.	CSIS End, min.	CSRS Start, min.	CSRS End, min.	Puff Release Time, min.	Puff Release Fraction, v/o[a]	Leakage Time Interval, min.	Leakage Leak Rate, v/o/hr[a]	Leakage Pressure, psia	Leakage Temperature F	Remarks
ACDGIδ	--	--	3.3	1200	1200	85	0-16	4.2×10^{-2}	56	268	Core heating, nominal leakage
							-60	4.2×10^{-2}	43	246	Core melting, nominal leakage
							-120	4.2×10^{-2}	16	150	Reactor vessel melting
							-140	4.2×10^{-3}	21	180	Boiloff and concrete decomposition
							-170	2.1×10^{-2}	26	206	Boiloff and concrete decomposition
							-1200	4.2×10^{-2}	100	315	Boiloff and concrete decomposition
							1200-	41	15	212	Boiloff and concrete decomposition
ACDGIε	--	--	3.3	1200	1200	66	0-16	4.2×10^{-2}	56	268	Core heating, nominal leakage
							-60	4.2×10^{-2}	43	246	Core melting, nominal leakage
							-120	4.2×10^{-2}	16	150	Reactor vessel melting
							-140	4.2×10^{-3}	21	180	Boiloff and concrete decomposition
							-170	2.1×10^{-2}	26	206	Boiloff and concrete decomposition
							-1200	4.2×10^{-2}	100	315	Boiloff and concrete decomposition
							1200-	4.2×10^{-2}	34	218	Pressure set by external head
ACDGIβ	--	--	3.3	--	--	--	0-16	42	50	259	Core heating, large leak
							-60	42	25	205	Core melting, large leak
							-120	21	15	212	Reactor vessel melting
							-140	42	25	240	Boiloff and decomposition
							140-	21	20	228	Boiloff and decomposition
ADα	0.5	94	3.3	--	60	8	0-16	4.2×10^{-2}	24	204	Core heating, nominal leakage
							-60	1.4×10^{-2}	15	150	Core melting
							-120	0	15	150	Reactor vessel melting
							-140	8.3	15	150	CO_2 leakage
							-180	1.6	15	150	CO_2 leakage
							-420	1.4	15	150	CO_2 leakage
							-900	1.1	15	150	CO_2 leakage
							900-	0.83	15	150	CO_2 leakage
ADε	0.5	94	3.3	--	--	--	0-16	4.2×10^{-2}	24	204	Core heating, nominal leakage
							-60	1.4×10^{-2}	15	150	Core melting
							-120	4.2×10^{-3}	16	150	Reactor vessel melting
							-140	8.0×10^{-3}	18	150	Concrete decomposition, CO_2 leakage
							140-350	2.1×10^{-2}	25	150	Concrete decomposition, CO_2 leakage
							-1200	4.2×10^{-2}	29	150	Concrete decomposition, CO_2 leakage
							1200-	4.2×10^{-2}	34	150	Pressure set by external head

TABLE 3 (CONTINUED)

Subsequence	CSIS Start, min.	CSIS End, min.	CSRS Start, min.	CSRS End, min.	Puff Release Time, min.	Puff Release Fraction, v/o[a]	Leakage Time Interval, min.	Leakage Leak Rate, v/o/hr[a]	Leakage Pressure, psia	Leakage Temperature F	Remarks
ADβ	0.5	94	3.3	--	--	--	0-16	42	24	204	Core heating, large leak
							-60	20	15	150	Core melting
							-120	0	15	150	Reactor vessel melting
							-140	17	15	150	Concrete decomposition, CO_2 leakage
							-180	3.2	15	150	Concrete decomposition, CO_2 leakage
							-420	2.8	15	150	Concrete decomposition, CO_2 leakage
							-900	2.2	15	150	Concrete decomposition, CO_2 leakage
							900-	1.6	15	150	Concrete decomposition, CO_2 leakage
ACDα	--	--	3.3	--	60	65	0-16	4.2×10^{-2}	56	268	Core heating, nominal leakage
							-60	4.2×10^{-2}	43	246	Core melting, nominal leakage
							-120	0	15	150	Reactor vessel melting
							-140	8.3	15	150	Concrete decomposition, CO_2 leakage
							-180	1.6	15	150	Concrete decomposition, CO_2 leakage
							-420	1.4	15	150	Concrete decomposition, CO_2 leakage
							-900	1.1	15	150	Concrete decomposition, CO_2 leakage
							900-	0.83	15	150	Concrete decomposition, CO_2 leakage
ACDε	--	--	3.3	--	--	--	0-16	4.2×10^{-2}	56	268	Core heating, nominal leakage
							-60	4.2×10^{-2}	43	246	Core melting, nominal leakage
							-120	4.2×10^{-2}	16	150	Reactor vessel melting
							-140	8.0×10^{-3}	18	150	Concrete decomposition, CO_2 leakage
							-350	2.1×10^{-2}	25	150	Concrete decomposition, CO_2 leakage
							-1200	4.2×10^{-2}	29	150	Concrete decomposition, CO_2 leakage
							1200-	4.2×10^{-2}	34	150	Pressure set by external head
ACDβ	--	--	3.3	--	--	--	0-16	42	50	259	Core heating, large leak
							-60	42	25	205	Core melting, large leak
							-120	21	15	150	Reactor vessel melting
							-140	17	15	150	Concrete decomposition, CO_2 leakage
							-180	3.2	15	150	Concrete decomposition, CO_2 leakage
							-420	2.8	15	150	Concrete decomposition, CO_2 leakage
							-900	2.2	15	150	Concrete decomposition, CO_2 leakage
							900-	1.6	15	150	Concrete decomposition, CO_2 leakage
AGHδ	0.5	47	3.3	1200	1280	85	0-30	4.2×10^{-2}	16	150	Nominal leakage, gap release
							-100	0	14	150	Core heating
							-150	4.2×10^{-3}	16	150	Core melting
							-210	4.2×10^{-3}	16	150	Reactor vessel melting

TABLE 3 (CONTINUED)

Subsequence	CSIS Start, min.	CSIS End, min.	CSRS Start, min.	CSRS End, min.	Puff Release Time, min.	Puff Release Fraction, v/o[a]	Leakage Time Interval, min.	Leakage Leak Rate, v/o/hr[A]	Leakage Pressure, psia	Leakage Temperature F	Remarks
							-320	2.1×10^{-2}	25	205	Boiloff and concrete decomposition
							-1280	4.2×10^{-2}	100	315	Boiloff and concrete decomposition
							1280-	41	15	212	Boiloff and concrete decomposition
AGHε	0.5	47	3.3	1290	1290	66	0-30	4.2×10^{-2}	16	150	Nominal leakage, gap release
							-100	0	14	150	Core heating
							-150	4.2×10^{-3}	16	150	Core melting
							-210	4.2×10^{-3}	16	150	Reactor vessel melting
							-320	2.1×10^{-2}	25	205	Boiloff and concrete decomposition
							-1290	4.2×10^{-2}	100	315	Boiloff and concrete decomposition
							1290-	4.2×10^{-2}	34	218	Pressure set by external head
AFHα	0.5	47	--	--	150	69	0-47	4.2×10^{-2}	20	174	Nominal leakage, gap release
							-60	2.1×10^{-2}	24	204	Pressure rises after sprays stop
							-100	4.2×10^{-2}	34	218	Core heating
							-150	4.2×10^{-2}	46	251	Core melting, nominal leakage
							-210	0	15	212	Reactor vessel melting
							-230	105	15	212	Boiloff and concrete decomposition
							230-	57	15	212	Boiloff and concrete decomposition
AFHδ	0.5	47	--	--	230	81	0-47	4.2×10^{-2}	20	174	Nominal leakage, gap release
							-60	2.1×10^{-2}	24	204	Pressure rises after sprays stop
							-100	4.2×10^{-2}	34	218	Core heating
							-150	4.2×10^{-2}	46	251	Core melting, nominal leakage
							-210	4.2×10^{-2}	40	240	Reactor vessel melting
							-230	4.2×10^{-2}	80	296	Boiloff and concrete decomposition
							-420	8.1	15	212	Concrete decomposition
							-900	6.0	15	212	Concrete decomposition
							-1290	5.1	15	212	Concrete decomposition
AFHε	0.5	47	--	--	1290	24	0-47	4.2×10^{-2}	20	174	Nominal leakage, gap release
							-60	2.1×10^{-2}	24	204	Pressure rises after sprays stop
							-100	4.2×10^{-2}	34	218	Core heating
							-150	4.2×10^{-2}	46	251	Core melting, nominal leakage
							150-210	4.2×10^{-2}	40	240	Reactor vessel melting
							-230	4.2×10^{-2}	80	296	Boiloff and concrete decomposition
							-1290	4.2×10^{-2}	45	250	Concrete decomposition
							1290-	4.2×10^{-2}	34	218	Pressure set by external head

TABLE 3 (CONTINUED)

Subsequence	CSIS Start, min.	CSIS End, min.	CSRS Start, min.	CSRS End, min.	Puff Release Time, min.	Puff Release Fraction, v/o[a]	Leakage Time Interval, min.	Leakage Leak Rate[d] v/o/hr	Leakage Pressure, psia	Leakage Temperature F	Remarks
AFHB	0.5	47	--	--	--	--	0-30	42	25	205	Large leak, gap release
							-45	0	15	212	Large leak
							-60	25	15	212	Sprays ended
							-100	32	15	212	Boiloff and core heating
							-150	32	15	212	Boiloff and core melting
							-210	42	30	250	Reactor vessel melting
							-230	42	30	250	Boiloff and concrete decomposition
							-420	42	15	212	Boiloff and concrete decomposition
							420-	36	15	212	Concrete decomposition
ADFc	0.5	94	--	--	1200	32	0-16	4.2×10^{-2}	41	240	Core heating, nominal leakage
							-40	4.2×10^{-2}	25	205	Core melting, nominal leakage
							-60	2.1×10^{-2}	16	150	Core melting
							-120	0	15	150	Reactor vessel melting
							-160	2.1×10^{-2}	25	205	Boiloff and concrete decomposition
							-1200	4.2×10^{-2}	50	259	Boiloff and concrete decomposition
							1200-	4.2×10^{-2}	34	218	Pressure set by external head
ADFB	0.5	94	--	--	--	--	0-16	42	40	240	Core heating, large leak
							-30	42	25	208	Core melting, large leak
							-60	21	15	212	Core melting, large leak
							-120	0	15	212	Reactor vessel melting
							-140	21	25	240	Boiloff and concrete decomposition
							-420	42	25	240	Boiloff and concrete decomposition
							-900	2.2	15	212	Concrete decomposition
							900-	1.6	15	212	Concrete decomposition
AHI Same as AH											
ADI Same as AD											
ADGc	0.5	94	3.3	1200	1200	66	0-16	4.2×10^{-2}	34	218	Core heating, nominal leakage
							-35	4.2×10^{-2}	25	208	Core melting, nominal leakage
							-60	2.1×10^{-2}	15	150	Core melting
							-120	0	15	150	Reactor vessel melting
							-160	2.1×10^{-2}	25	208	Boiloff and concrete decomposition
							-1200	4.2×10^{-2}	100	318	Boiloff and concrete decomposition
							1200-	4.2×10^{-2}	34	218	Pressure set by external head
ADGI Same as ADG											

TABLE 3 (CONTINUED)

Subsequence	CSIS Start, min.	CSIS End, min.	CSRS Start, min.	CSRS End, min.	Puff Release Time, min.	Puff Release Fraction, v/o[a]	Leakage Time Interval, min.	Leakage Leak Rate, v/o/hr[b]	Leakage Pressure, psia	Leakage Temperature F	Remarks
$S_2C\alpha$	—	—	—	—			0-10	2.8×10^{-2}	20	190	Nominal leakage
							-50	4.2×10^{-2}	39	244	Nominal leakage
							-100	4.2×10^{-2}	58	275	Nominal leakage
							-240	4.2×10^{-2}	100	315	Nominal leakage
					240	85	240	—	15	212	Containment failure
							240-290	55	15	212	Boiloff and core heatup
							290-360	48	15	212	Core melt
					360	28	360	—	15	212	Steam explosion
							360-440	—	15	212	Reactor vessel melt
							440-460	25	15	212	Initial concrete attack
							460-900	4	15	212	CO_2 and H_2O from concrete
							900-1800	3	15	212	CO_2 and H_2O from concrete
							1800-	2	15	212	CO_2 and H_2O from concrete
$S_2C\delta$	—	—	—	—			0-10	2.8×10^{-2}	20	190	Nominal leakage
							-50	4.2×10^{-2}	39	244	Nominal leakage
							-100	4.2×10^{-2}	58	275	Nominal leakage
							-240	4.2×10^{-2}	100	315	Nominal leakage
					240	85	240	—	15	212	Containment failure
							240-290	55	15	212	Boiloff and core heatup
							290-300	48	15	212	Core melt
							360-440	—	15	212	Reactor vessel melt
							440-460	50	15	212	Initial Concrete Attack
							460-900	8	15	212	Initial Concrete Attack
							900-1800	6	15	212	Initial Concrete Attack
							1800-	4	15	212	Initial Concrete Attack
TMLB'γ	—	—	—	—			0-10	4.2×10^{-2}	59	276	Nominal leakage
							-50	4.2×10^{-2}	69	289	Nominal leakage
							-100	4.2×10^{-2}	80	299	Nominal leakage
							-150	4.2×10^{-2}	100	315	Core heatup, nominal leakage
							-220	4.2×10^{-2}	130	343	Core melt, nominal leakage
					220	88	220	—	15	212	Containment failure
							220-280	—	15	212	Reactor vessel melt
							280-300	50	15	212	Initial concrete attack
							300-420	9	15	212	CO_2 and H_2O from concrete
							-900	6	15	212	CO_2 and H_2O from concrete
							900-	5	15	212	CO_2 and H_2O from concrete

TABLE 3 (CONTINUED)

Subsequence	CSIS Start, min.	CSIS End, min.	CSRS Start, min.	CSRS End, min.	Puff Release Time, min.	Puff Release Fraction, v/o[a]	Leakage Time Interval, min.	Leakage Leak Rate[a] v/o/hr	Leakage Pressure, psia	Leakage Temperature F	Remarks
TMLB'δ	—	—	—	—			0-10	$4.2x10^{-2}$	59	276	Nominal leakage
							-50	$4.2x10^{-2}$	69	289	Nominal leakage
							-100	$4.2x10^{-2}$	80	299	Nominal leakage
							-150	$4.2x10^{-2}$	90	306	Core heatup, nominal leakage
							-220	$4.2x10^{-2}$	105	319	Core melt, nominal leakage
					220	86	220	—	15	212	Containment failure
							220-280	—	15	212	Reactor vessel melt
							280-300	50	15	212	Initial concrete attack
							300-420	9	15	212	CO_2 and H_2O from concrete
							-900	6	15	212	CO_2 and H_2O from concrete
							900-	5	15	212	CO_2 and H_2O from concrete
TXLB'α	—	—	—	—			0-10	$4.2x10^{-2}$	59	276	Nominal leakage
							-50	$4.2x10^{-2}$	69	289	Nominal leakage
							-100	$4.2x10^{-2}$	80	299	Nominal leakage
							-150	$4.2x10^{-2}$	90	306	Core heatup, nominal leakage
							-220	$4.2x10^{-2}$	105	319	Core melt, nominal leakage
					220	86	220	—	15	212	Containment failure
							220-280	—	15	212	Reactor vessel melt
							280-300	25	15	212	Initial concrete attack
							300-420	5	15	212	CO_2 and H_2O from concrete
							420-900	3	15	212	CO_2 and H_2O from concrete
							900-	2	15	212	CO_2 and H_2O from concrete

(a) Normalized to a containment free volume of $1.8x10^6$ ft^3.

TABLE 4 FISSION PRODUCT RELEASE VALUES FOR CORRAL-PWR CALCULATIONS

Sequence	Release Component[a]	Time of Release, minutes	Fraction of Core Inventory Released to Containment[b]						
			Xe	I	Cs	Te	Sr	Ru	La
A	Gap	1	.030	.017	.050	10^{-4}	10^{-6}	--	--
Aβ	Gap	1	.030	.017	.050	10^{-4}	10^{-6}	--	--
AHα	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	100-150	.870	.860	.700	.150	.100	.030	.003
	St. Expl.	150	.045	.045	--	.255	--	.436	--
	Vaporiz	210-330	.050	.050	.085	.425	.005	.025	.005
AHβ	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	100-150	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	210-330	.100	.100	.170	.850	.010	.050	.010
AHε	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	100-150	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	210-330	.100	.100	.170	.850	.010	.050	.010
AHIα	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	100-150	.870	.860	.700	.150	.100	.030	.003
	St. Expl.	150	.045	.045	--	.255	--	.436	--
	Vaporiz	210-330	.050	.050	.085	.425	.005	.025	.005
AHIβ	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	100-150	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	210-330	.100	.100	.170	.850	.010	.050	.010
AHIε	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	100-150	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	210-330	.100	.100	.170	.850	.010	.050	.010
AGδ	Gap	Ignore to simplify calculation							
	Melt	1370-1490	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	1610-1850	.100	.100	.170	.850	.010	.050	.010
AHGδ	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	100-150	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	210-330	.100	.100	.170	.850	.010	.100	.005
AHGε	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	100-150	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	210-330	.100	.100	.170	.850	.010	.100	.005
AHFα	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	100-150	.870	.860	.700	.150	.100	.030	.003
	St. Expl.	150	.045	.045	--	.255	--	.436	--
	Vaporiz	210-330	.100	.100	.170	.425	.005	.025	.005

TABLE 4 (CONTINUED)

Sequence	Release Component[a]	Time of Release, minutes	Fraction of Core Inventory Released to Containment[b]						
			Xe	I	Cs	Te	Sr	Ru	La
AHFβ	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	100-150	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	210-330	.100	.100	.170	.850	.010	.050	.010
AHFδ	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	100-150	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	210-330	.100	.100	.170	.850	.010	.050	.010
AHFε	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	100-150	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	210-330	.100	.100	.170	.850	.010	.050	.010
ADα	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	St. Expl.	60	.045	.045	--	.255	--	.436	--
	Vaporiz	120-240	.050	.050	.085	.425	.005	.025	.005
ADβ	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	120-140	.100	.100	.170	.850	.010	.050	.010
ADε	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	120-240	.100	.100	.170	.850	.010	.050	.010
ADIα	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	St. Expl.	60	.045	.045	--	.255	--	.436	--
	Vaporiz	120-240	.050	.050	.085	.425	.005	.025	.005
ADIε	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	120-240	.100	.100	.170	.850	.010	.050	.010
ADGε	Gap	1	.030	.017	.050	10^{-4}	10^{-6}	--	--
	Melt	16-60	.870	.883	.760	.150	.100	.030	.003
	Vaporiz	120-240	.100	.100	.190	.850	.010	.050	.010
ADGIε	Gap	1	.030	.017	.050	10^{-4}	10^{-6}	--	--
	Melt	16-60	.870	.883	.760	.150	.100	.030	.003
	Vaporiz	120-240	.100	.100	.190	.850	.010	.050	.010
ADFβ	Gap	1	.030	.040	.130	.001	.000015	--	--
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	120-240	.100	.100	.170	.850	.010	.050	.010

TABLE 5 CONSTANTS AND PARAMETERS USED IN CORRAL-PWR RUNS

	Reactor Compartment Data			
Compartment	Wall Area, ft^2	Floor Area, ft^2	Height, ft	Volume, ft^3
Main Volume	4.06×10^4	8.83×10^3	1.1×10^2	9.4×10^5
Primary Cubicle	7.3×10^3	1.2×10^3	4.9×10^1	6.0×10^4
Outer Annulus	8.75×10^4	3.75×10^3	1.2×10^2	4.5×10^5
Lower Volume	1.66×10^4	7.4×10^3	2.2×10^1	1.6×10^5

Containment Spray Parameters

Parameter	Value
CSIS flow rate, ft^3/hr	2.57×10^4
CSRS flow rate (1 pump), ft^3/hr	2.81×10^4
CSRS flow rate (2 pumps), ft^3/hr	5.62×10^4
CSIS spray drop fall height, ft	100
CSRS spray drop fall height, ft	47
CSIS spray drop diameter, cm	0.10
CSRS spray drop diameter, cm	0.11
Equilibrium I_2 partition coefficient	5000
Iodine C/Co limit for spray removal	0.01

Other Parameters

Aerosol particle diameter, μm
- Early 15
- Late 5

Time interval of aerosol particle diameter change = 4 hr.

Organic iodine conversion ratio
- If sprays operate 0.4%
- If no sprays operate 0.7%

Flow rate between compartments = 10 volume/hr.

Decomtamination factor between compartments = 1 (all species)

Decontamination factor for leaks to the atmosphere = 1 (all species)

Decontamination factor for leakage through the ground = 1 (noble gases and organic iodine) and 1000 (all other species).

TABLE 4 (CONTINUED)

Sequence	Release Component[a]	Time of Release, minutes	Fraction of Core Inventory Released to Containment[b]						
			Xe	I	Cs	Te	Sr	Ru	La
ADFε	Gap	1	.030	.040	.130	.001	.000015	—	—
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	120-240	.100	.100	.170	.850	.010	.050	.003
ACDβ	Gap	1	.030	.040	.130	.001	.000015	—	—
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	120-240	.100	.100	.170	.850	.010	.050	.010
ACDε	Gap	1	.030	.040	.130	.001	.000015	—	—
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	120-240	.100	.100	.170	.850	.010	.050	.010
ACDGIα	Gap	1	.030	.040	.130	.001	.000015	—	—
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	St. Expl.	60	.045	.045	—	.255	—	.436	—
	Vaporiz	120-240	.050	.050	.085	.425	.005	.025	.005
ACDGIβ	Gap	1	.030	.040	.130	.001	.000015	—	—
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	120-240	.100	.100	.170	.850	.010	.050	.010
ACDGIδ	Gap	1	.030	.040	.130	.001	.000015	—	—
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	120-240	.100	.100	.170	.850	.010	.050	.010
ACDGIε	Gap	1	.030	.040	.130	.001	.000015	—	—
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	120-240	.100	.100	.170	.850	.010	.050	.010
ABα	Gap	1	.030	.040	.130	.001	.000015	—	—
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	St. Expl.	60	.045	.045	—	.255	—	.436	—
	Vaporiz	120-240	.050	.050	.085	.425	.005	.025	.005
ABγ	Gap	1	.030	.040	.130	.001	.000015	—	—
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	120-240	.100	.100	.170	.850	.010	.050	.010
ABε	Gap	1	.030	.040	.130	.001	.000015	—	—
	Melt	16-60	.870	.860	.700	.150	.100	.030	.003
	Vaporiz	120-240	.100	.100	.170	.850	.010	.050	.010
$S_2C\delta$	Gap	241	.030	.017	.050	10^{-4}	10^{-6}	0	0
	Melt	290-360	.870	.883	.760	.150	.100	.030	.003
	Vaporiz	440-560	.100	.100	.190	.850	.010	.050	.010

TABLE 4 (CONTINUED)

Sequence	Release Component[a]	Time of Release, minutes	Fraction of Core Inventory Released to Containment[b]						
			Xe	I	Cs	Te	Sr	Ru	La
$S_2C\alpha$	Gap	241	.030	.017	.050	10^{-4}	10^{-6}	0	0
	Melt	290-360	.870	.883	.760	.150	.100	.030	.003
	St. Expl.	360	.045	.045	—	.255	—	.437	—
	Vaporiz	440-560	.050	.050	.095	.425	.005	.024	.005
TMLB'γ	Gap	120	.030	.017	.050	10^{-4}	10^{-6}	0	0
	Melt	170-220	.870	.883	.760	.150	.100	.030	.003
	Vaporiz	280-400	.100	.100	.190	.850	.010	.050	.010
TMLB'δ	Gap	120	.030	.017	.050	10^{-4}	10^{-6}	0	0
	Melt	170-220	.870	.883	.760	.150	.100	.030	.003
	Vaporiz	280-400	.100	.100	.190	.850	.010	.050	.010
TMLB'α	Gap	120	.030	.017	.050	10^{-4}	10^{-6}	0	0
	Melt	170-220	.870	.883	.760	.150	.100	.030	.003
	St. Expl.	220	.045	.045	—	.255	—	.437	—
	Vaporiz	280-400	.050	.050	.095	.425	.005	.024	.005

(a) Gap means the gap release component
Melt means the core melt release component
St. Expl. means the steam explosion release component
Vaporiz. means the vaporization release component.

(b) XE also includes Kr
I also includes Br
Cs also includes Rb
Te also includes Se and Sb
Sr also includes Ba
Ru also includes Mo, Pd, Rh, and Tc
La also includes Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, and Nb.

TABLE 6 RESULTS OF CORRAL-PWR CALCULATIONS

Time, hr	Cumulative Fractions of Core Inventory Released to the Atmosphere[a]								Event[b]
	Xe-Kr	Org-I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La	
					Case A				
0.2	$1x10^{-6}$	$3x10^{-9}$	$1x10^{-7}$	$4x10^{-7}$	$8x10^{-10}$	$8x10^{-12}$	0	0	
0.5	$3x10^{-6}$	$7x10^{-9}$	$1x10^{-7}$	$6x10^{-7}$	$1x10^{-9}$	$1x10^{-11}$	0	0	End Leak
					Case Aβ				
0.2	$1x10^{-3}$	$2x10^{-6}$	$1x10^{-4}$	$4x10^{-4}$	$8x10^{-7}$	$8x10^{-9}$	0	0	
0.5	$2x10^{-3}$	$5x10^{-6}$	$1x10^{-4}$	$5x10^{-4}$	$1x10^{-6}$	$1x10^{-8}$	0	0	End Leak
					Case AHα				
1.6	$3x10^{-6}$	0	$2x10^{-7}$	$1x10^{-6}$	$8x10^{-9}$	$1x10^{-10}$	0	0	Just Before Melt
2.5	$2x10^{-5}$	$7x10^{-8}$	$1x10^{-6}$	$3x10^{-6}$	$4x10^{-7}$	$3x10^{-7}$	$8x10^{-8}$	$8x10^{-9}$	End Melt Rel.
2.5	$6x10^{-2}$	$3x10^{-4}$	$6x10^{-3}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$3x10^{-2}$	$2x10^{-5}$	Steam Expl.
4.0	$9x10^{-2}$	$4x10^{-4}$	$6x10^{-3}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$3x10^{-2}$	$3x10^{-5}$	
5.4	$1x10^{-1}$	$4x10^{-4}$	$6x10^{-3}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$3x10^{-2}$	$3x10^{-5}$	End Vap. Rel.
17	$2x10^{-1}$	$8x10^{-4}$	$7x10^{-3}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$3x10^{-2}$	$4x10^{-5}$	
720	1.0	$4x10^{-3}$	$1x10^{-2}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$3x10^{-2}$	$4x10^{-5}$	
					Case AHβ				
1.7	$5x10^{-3}$	0	$7x10^{-5}$	$5x10^{-4}$	$4x10^{-6}$	$6x10^{-8}$	0	0	Just Before Melt
2.5	$6x10^{-2}$	$1x10^{-4}$	$4x10^{-3}$	$8x10^{-3}$	$2x10^{-3}$	$1x10^{-3}$	$3x10^{-4}$	$3x10^{-5}$	End Melt Rel.
4.0	$1x10^{-1}$	$5x10^{-4}$	$5x10^{-3}$	$9x10^{-3}$	$4x10^{-3}$	$1x10^{-3}$	$5x10^{-4}$	$6x10^{-5}$	
5.5	$1x10^{-1}$	$6x10^{-4}$	$5x10^{-3}$	$1x10^{-2}$	$5x10^{-3}$	$1x10^{-3}$	$5x10^{-4}$	$7x10^{-5}$	End Vap. Rel.
15	$3x10^{-1}$	$1x10^{-3}$	$1x10^{-3}$	$1x10^{-2}$	$5x10^{-3}$	$1x10^{-3}$	$5x10^{-4}$	$7x10^{-5}$	
710	1.0	$4x10^{-3}$	$8x10^{-3}$	$1x10^{-2}$	$5x10^{-3}$	$1x10^{-3}$	$5x10^{-4}$	$7x10^{-5}$	
					Case AHε				
1.7	$4x10^{-6}$	0	$4x10^{-7}$	$1x10^{-6}$	$1x10^{-8}$	$2x10^{-10}$	0	0	Just Before Melt
2.5	$2x10^{-5}$	$7x10^{-8}$	$1x10^{-6}$	$3x10^{-6}$	$5x10^{-7}$	$3x10^{-7}$	$9x10^{-8}$	$9x10^{-9}$	End Melt Rel.
4.0	$1x10^{-4}$	$4x10^{-7}$	$2x10^{-6}$	$6x10^{-6}$	$2x10^{-6}$	$6x10^{-7}$	$3x10^{-7}$	$3x10^{-8}$	
5.5	$2x10^{-4}$	$9x10^{-7}$	$2x10^{-6}$	$7x10^{-6}$	$5x10^{-6}$	$7x10^{-7}$	$4x10^{-7}$	$5x10^{-8}$	End Vap. Rel.
15	$2x10^{-3}$	$9x10^{-6}$	$7x10^{-6}$	$7x10^{-6}$	$6x10^{-6}$	$7x10^{-7}$	$5x10^{-7}$	$8x10^{-8}$	
720	$3x10^{-1}$	$1x10^{-3}$	$5x10^{-4}$	$7x10^{-6}$	$6x10^{-6}$	$7x10^{-7}$	$5x10^{-7}$	$8x10^{-8}$	
					Case AHIα				
1.6	$3x10^{-6}$	0	$1x10^{-6}$	$1x10^{-6}$	$8x10^{-9}$	$1x10^{-10}$	0	0	Just Before Melt
2.5	$2x10^{-5}$	$7x10^{-8}$	$9x10^{-6}$	$3x10^{-6}$	$4x10^{-7}$	$3x10^{-7}$	$8x10^{-8}$	$8x10^{-9}$	End Melt Rel.
2.5	$6x10^{-2}$	$3x10^{-4}$	$2x10^{-2}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$3x10^{-2}$	$2x10^{-5}$	Steam Expl.
4.0	$9x10^{-2}$	$4x10^{-4}$	$3x10^{-2}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$3x10^{-2}$	$3x10^{-5}$	
5.4	$1x10^{-1}$	$4x10^{-4}$	$3x10^{-2}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$3x10^{-2}$	$3x10^{-5}$	End Vap. Rel.
17	$2x10^{-1}$	$8x10^{-4}$	$3x10^{-2}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$3x10^{-2}$	$4x10^{-5}$	
720	1.0	$4x10^{-3}$	$3x10^{-2}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$3x10^{-2}$	$4x10^{-5}$	

TABLE 6 (CONTINUED)

Time, hr	Cumulative Fractions of Core Inventory Released to the Atmosphere[a]								Event[b]
	Xe-Kr	Org-I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La	
					Case AHIβ				
1.7	5×10^{-3}	0	6×10^{-4}	5×10^{-4}	4×10^{-6}	6×10^{-8}	0	0	Just Before Melt
2.5	6×10^{-2}	3×10^{-4}	3×10^{-2}	8×10^{-3}	2×10^{-3}	1×10^{-3}	3×10^{-4}	3×10^{-5}	End Melt Rel.
4.0	1×10^{-1}	5×10^{-4}	3×10^{-2}	9×10^{-3}	4×10^{-3}	1×10^{-3}	5×10^{-4}	6×10^{-5}	
5.5	1×10^{-1}	6×10^{-4}	3×10^{-2}	1×10^{-2}	5×10^{-3}	1×10^{-3}	5×10^{-4}	7×10^{-5}	End Vap. Rel.
15	3×10^{-1}	1×10^{-3}	3×10^{-2}	1×10^{-2}	5×10^{-3}	1×10^{-3}	5×10^{-4}	7×10^{-5}	
720	1.0	4×10^{-3}	4×10^{-2}	1×10^{-2}	5×10^{-3}	1×10^{-3}	5×10^{-4}	7×10^{-5}	
					Case AHIε				
1.7	4×10^{-6}	0	2×10^{-6}	1×10^{-6}	1×10^{-8}	2×10^{-10}	0	0	Just Before Melt
2.5	2×10^{-5}	7×10^{-8}	8×10^{-6}	3×10^{-6}	5×10^{-9}	3×10^{-7}	9×10^{-8}	9×10^{-9}	End Melt Rel.
4.0	1×10^{-4}	4×10^{-7}	1×10^{-5}	6×10^{-6}	2×10^{-6}	6×10^{-7}	3×10^{-7}	3×10^{-8}	
5.5	2×10^{-4}	9×10^{-7}	1×10^{-5}	7×10^{-6}	5×10^{-6}	7×10^{-7}	4×10^{-7}	7×10^{-8}	End Vap. Rel.
15	2×10^{-3}	9×10^{-6}	2×10^{-5}	7×10^{-6}	6×10^{-6}	7×10^{-7}	5×10^{-7}	8×10^{-8}	
720	3×10^{-1}	1×10^{-3}	5×10^{-4}	7×10^{-6}	6×10^{-6}	7×10^{-7}	5×10^{-7}	8×10^{-8}	
					Case AGδ				
23.8	5×10^{-2}	5×10^{-4}	4×10^{-2}	3×10^{-2}	6×10^{-3}	4×10^{-3}	1×10^{-3}	1×10^{-4}	1/2 Melt Rel.
26.8	2×10^{-1}	2×10^{-3}	1×10^{-1}	8×10^{-2}	2×10^{-2}	1×10^{-2}	3×10^{-3}	3×10^{-4}	PV Meltthrough
27.8	4×10^{-1}	3×10^{-3}	2×10^{-1}	1×10^{-1}	8×10^{-2}	1×10^{-2}	7×10^{-3}	1×10^{-3}	1/2 Vap. Rel.
30.9	8×10^{-1}	6×10^{-3}	2×10^{-1}	2×10^{-1}	2×10^{-1}	2×10^{-2}	2×10^{-2}	3×10^{-3}	End Vap. Rel.
49.2	9×10^{-1}	7×10^{-3}	2×10^{-1}	2×10^{-1}	3×10^{-1}	2×10^{-2}	2×10^{-2}	3×10^{-3}	
720	9×10^{-1}	7×10^{-3}	2×10^{-1}	2×10^{-1}	3×10^{-1}	2×10^{-2}	2×10^{-2}	3×10^{-3}	
					Case AHGδ				
1.7	6×10^{-6}	0	2×10^{-7}	2×10^{-6}	2×10^{-8}	2×10^{-10}	0	0	Just Before Melt
2.5	2×10^{-5}	7×10^{-8}	1×10^{-6}	4×10^{-6}	4×10^{-7}	3×10^{-7}	9×10^{-8}	9×10^{-9}	End Melt Rel.
5.5	5×10^{-4}	2×10^{-6}	4×10^{-6}	1×10^{-5}	1×10^{-5}	9×10^{-7}	8×10^{-7}	1×10^{-7}	End Vap. Rel.
19.9	5×10^{-3}	2×10^{-5}	2×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}	
20.0	9×10^{-1}	4×10^{-3}	2×10^{-3}	1×10^{-5}	1×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}	Overpressure
720	1.0	4×20^{-3}	2×10^{-3}	1×10^{-5}	1×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}	
					Case AHGε				
1.7	6×10^{-6}	0	2×10^{-7}	2×10^{-6}	2×10^{-8}	2×10^{-10}	0	0	Just Before Melt
2.5	2×10^{-5}	7×10^{-8}	1×10^{-6}	4×10^{-6}	4×10^{-7}	3×10^{-7}	9×10^{-8}	9×10^{-9}	End Melt Rel.
5.5	5×10^{-4}	2×10^{-6}	4×10^{-6}	1×10^{-5}	1×10^{-5}	9×10^{-7}	8×10^{-7}	1×10^{-7}	End Vap. Rel.
21.1	7×10^{-3}	3×10^{-5}	2×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}	
21.5	7×10^{-1}	3×10^{-3}	2×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}	Cont. Meltthrough
720	7×10^{-1}	3×10^{-3}	2×10^{-4}	1×10^{-5}	1×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}	

TABLE 6 (CONTINUED)

Time, hr	Cumulative Fractions of Core Inventory Released to the Atmosphere[a]								Event[b]
	Xe-Kr	Org-I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La	
					Case ADε				
0.2	$2x10^{-6}$	0	$4x10^{-7}$	$1x10^{-6}$	$8x10^{-9}$	$1x10^{-10}$	0	0	Just Before Melt
1.0	$5x10^{-5}$	$2x10^{-7}$	$2x10^{-6}$	$6x10^{-6}$	$9x10^{-7}$	$6x10^{-7}$	$2x10^{-7}$	$2x10^{-8}$	End Melt Rel.
2.5	$2x10^{-4}$	$1x10^{-6}$	$3x10^{-6}$	$8x10^{-6}$	$5x10^{-6}$	$9x10^{-7}$	$5x10^{-7}$	$7x10^{-8}$	
4.0	$5x10^{-4}$	$2x10^{-6}$	$5x10^{-6}$	$1x10^{-5}$	$1x10^{-5}$	$1x10^{-6}$	$1x10^{-6}$	$2x10^{-7}$	End Vap. Rel.
16.0	$6x10^{-3}$	$2x10^{-5}$	$2x10^{-5}$	$1x10^{-5}$	$2x10^{-5}$	$1x10^{-6}$	$1x10^{-6}$	$2x10^{-7}$	
720	$3x10^{-1}$	$1x10^{-3}$	$5x10^{-4}$	$1x10^{-5}$	$2x10^{-5}$	$1x10^{-6}$	$1x10^{-6}$	$2x10^{-7}$	
					Case ADIα				
0.2	$2x10^{-6}$	0	$2x10^{-6}$	$1x10^{-6}$	$1x10^{-8}$	$1x10^{-10}$	0	0	Just Before Melt
1.0	$4x10^{-5}$	$2x10^{-7}$	$2x10^{-5}$	$5x10^{-6}$	$7x10^{-7}$	$5x10^{-7}$	$2x10^{-7}$	$2x10^{-8}$	End Melt Rel.
1.0	$8x10^{-2}$	$3x10^{-4}$	$2x10^{-2}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$4x10^{-2}$	$2x10^{-5}$	Steam Expl.
2.5	$1x10^{-1}$	$4x10^{-4}$	$2x10^{-2}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$4x10^{-2}$	$3x10^{-5}$	
4.0	$1x10^{-1}$	$5x10^{-4}$	$2x10^{-2}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$4x10^{-2}$	$3x10^{-5}$	End Vap. Rel.
16.0	$2x10^{-1}$	$9x10^{-4}$	$2x10^{-2}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$4x10^{-2}$	$3x10^{-5}$	
720	1.0	$4x10^{-3}$	$3x10^{-2}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$4x10^{-2}$	$3x10^{-5}$	
					Case ADIε				
0.26	$2x10^{-6}$	0	$1x10^{-6}$	$1x10^{-6}$	$8x10^{-9}$	$1x10^{-10}$	0	0	Just Before Melt
1.0	$5x10^{-5}$	$2x10^{-5}$	$2x10^{-7}$	$6x10^{-6}$	$9x10^{-7}$	$6x10^{-7}$	$2x10^{-7}$	$2x10^{-8}$	End Melt Rel.
2.5	$2x10^{-4}$	$1x10^{-6}$	$8x10^{-6}$	$2x10^{-5}$	$8x10^{-6}$	$9x10^{-7}$	$5x10^{-7}$	$7x10^{-8}$	
4.0	$5x10^{-4}$	$2x10^{-6}$	$3x10^{-5}$	$1x10^{-5}$	$1x10^{-5}$	$1x10^{-6}$	$1x10^{-6}$	$2x10^{-7}$	End Vap. Rel.
16.0	$6x10^{-3}$	$2x10^{-5}$	$4x10^{-5}$	$1x10^{-5}$	$2x10^{-5}$	$1x10^{-6}$	$1x10^{-6}$	$2x10^{-7}$	
720	$3x10^{-1}$	$1x10^{-3}$	$5x10^{-4}$	$1x10^{-5}$	$2x10^{-5}$	$1x10^{-6}$	$1x10^{-6}$	$2x10^{-7}$	
					Case ADGε				
0.26	$2x10^{-6}$	0	$2x10^{-7}$	$5x10^{-7}$	$9x10^{-7}$	$9x10^{-12}$	0	0	Just Before Melt
1.0	$8x10^{-5}$	$3x10^{-7}$	$4x10^{-6}$	$9x10^{-6}$	$2x10^{-6}$	$1x10^{-6}$	$3x10^{-7}$	$3x10^{-8}$	End Melt Rel.
2.5	$3x10^{-4}$	$1x10^{-6}$	$5x10^{-6}$	$1x10^{-5}$	$7x10^{-6}$	$1x10^{-6}$	$6x10^{-7}$	$9x10^{-8}$	
4.0	$9x10^{-4}$	$4x10^{-6}$	$9x10^{-6}$	$2x10^{-5}$	$2x10^{-5}$	$2x10^{-6}$	$2x10^{-6}$	$3x10^{-7}$	End Vap. Rel.
20.0	$7x10^{-1}$	$3x10^{-3}$	$4x10^{-5}$	$2x10^{-5}$	$3x10^{-5}$	$2x10^{-6}$	$2x10^{-6}$	$3x10^{-7}$	Cont. Meltthrough
720	$7x10^{-1}$	$3x10^{-3}$	$3x10^{-4}$	$2x10^{-5}$	$3x10^{-5}$	$2x10^{-6}$	$2x10^{-6}$	$3x10^{-7}$	
					Case ADGIε				
0.26	$2x10^{-6}$	0	$7x10^{-7}$	$5x10^{-7}$	$9x10^{-10}$	$9x10^{-12}$	0	0	Just Before Melt
1.0	$8x10^{-5}$	$3x10^{-7}$	$3x10^{-5}$	$9x10^{-6}$	$2x10^{-6}$	$1x10^{-6}$	$3x10^{-7}$	$3x10^{-8}$	End Melt Rel.
2.5	$3x10^{-4}$	$1x10^{-6}$	$4x10^{-5}$	$1x10^{-5}$	$6x10^{-6}$	$1x10^{-6}$	$6x10^{-7}$	$9x10^{-8}$	
4.0	$9x10^{-4}$	$4x10^{-6}$	$4x10^{-5}$	$2x10^{-5}$	$2x10^{-5}$	$2x10^{-6}$	$2x10^{-6}$	$3x10^{-7}$	End Vap. Rel.
20.0	$7x10^{-1}$	$3x10^{-3}$	$6x10^{-5}$	$2x10^{-5}$	$3x10^{-5}$	$2x10^{-6}$	$2x10^{-6}$	$3x10^{-7}$	Cont. Meltthrough
720	$7x10^{-1}$	$3x10^{-3}$	$2x10^{-4}$	$2x10^{-5}$	$3x10^{-5}$	$2x10^{-6}$	$2x10^{-6}$	$3x10^{-7}$	

TABLE 6 (CONTINUED)

Time, hr	Cumulative Fractions of Core Inventory Released to the Atmosphere[a]								Event[b]
	Xe-Kr	Org-I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La	
					Case AHFα				
1.6	$2x10^{-5}$	0	$2x10^{-6}$	$2x10^{-6}$	$2x10^{-8}$	$3x10^{-10}$	0	0	Just Before Melt
2.5	$2x10^{-4}$	$1x10^{-6}$	$1x10^{-4}$	$8x10^{-5}$	$2x10^{-5}$	$1x10^{-5}$	$3x10^{-6}$	$3x10^{-7}$	End Melt Rel.
2.5	$7x10^{-1}$	$5x10^{-3}$	$5x10^{-1}$	$3x10^{-1}$	$2x10^{-1}$	$4x10^{-2}$	$3x10^{-1}$	$1x10^{-3}$	Steam Expl.
5.5	$9x10^{-1}$	$6x10^{-3}$	$6x10^{-1}$	$3x10^{-1}$	$4x10^{-1}$	$5x10^{-2}$	$3x10^{-1}$	$3x10^{-3}$	End Vap. Rel.
720	1.0	$7x10^{-3}$	$6x10^{-1}$	$4x10^{-1}$	$4x10^{-1}$	$5x10^{-2}$	$3x10^{-1}$	$3x10^{-3}$	
					Case AHFδ				
1.6	$2x10^{-5}$	0	$2x10^{-6}$	$2x10^{-6}$	$2x10^{-8}$	$3x10^{-10}$	0	0	Just Before Melt
3.8	$7x10^{-4}$	$5x10^{-6}$	$4x10^{-4}$	$2x10^{-4}$	$7x10^{-5}$	$3x10^{-5}$	$1x10^{-5}$	$1x10^{-6}$	Just Before OP
3.8	$8x10^{-1}$	$6x10^{-3}$	$3x10^{-1}$	$2x10^{-1}$	$3x10^{-1}$	$2x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	Overpressure
5.5	$8x10^{-1}$	$6x10^{-3}$	$3x10^{-1}$	$2x10^{-1}$	$3x10^{-1}$	$2x10^{-2}$	$2x10^{-2}$	$4x10^{-3}$	End Vap. Rel.
720	1.0	$7x10^{-3}$	$4x10^{-1}$	$2x10^{-1}$	$3x10^{-1}$	$2x10^{-2}$	$2x10^{-2}$	$4x10^{-3}$	
					Case AHFβ				
.6	$1x10^{-2}$	0	$1x10^{-3}$	$9x10^{-5}$	$7x10^{-7}$	$1x10^{-8}$	0	0	Just Before Melt
2.5	$2x10^{-1}$	$1x10^{-3}$	$1x10^{-1}$	$9x10^{-2}$	$2x10^{-2}$	$1x10^{-2}$	$4x10^{-3}$	$4x10^{-4}$	End Melt Rel.
5.5	$8x10^{-1}$	$6x10^{-3}$	$4x10^{-1}$	$3x10^{-1}$	$3x10^{-1}$	$4x1-^{-2}$	$2x10^{-2}$	$3x10^{-3}$	End Vap. Rel.
720	1.0	$7x10^{-3}$	$4x10^{-1}$	$3x10^{-1}$	$3x10^{-1}$	$4x10^{-2}$	$3x10^{-2}$	$4x10^{-3}$	
					Case AHFε				
1.6	$2x10^{-5}$	0	$2x10^{-6}$	$2x10^{-6}$	$2x10^{-8}$	$3x10^{-10}$	0	0	Just Before Melt
2.5	$2x10^{-4}$	$1x10^{-6}$	$1x10^{-4}$	$9x10^{-5}$	$2x10^{-5}$	$1x10^{-5}$	$4x10^{-6}$	$4x10^{-7}$	End Melt Rel.
5.5	$1x10^{-3}$	$1x10^{-5}$	$6x10^{-4}$	$4x10^{-4}$	$3x10^{-4}$	$5x10^{-5}$	$3x10^{-5}$	$4x10^{-6}$	End Vap. Rel.
20.6	$8x10^{-3}$	$5x10^{-5}$	$8x10^{-4}$	$7x10^{-4}$	$1x10^{-3}$	$9x10^{-5}$	$7x10^{-5}$	$1x10^{-5}$	
21.5	$2x10^{-1}$	$2x10^{-3}$	$8x10^{-4}$	$7x10^{-4}$	$1x10^{-3}$	$9x10^{-5}$	$8x10^{-5}$	$1x10^{-5}$	Cont. Meltthrough
720	$4x10^{-1}$	$3x10^{-3}$	$2x10^{-3}$	$8x10^{-4}$	$1x10^{-3}$	$9x10^{-5}$	$8x10^{-5}$	$1x10^{-5}$	
					Case ADα				
0.2	2.10^{-6}	0	$5x10^{-7}$	$1x10^{-6}$	$1x10^{-8}$	$1x10^{-10}$	0	0	Just Before Melt
1.0	$4x10^{-5}$	$2x10^{-7}$	$2x10^{-6}$	$5x10^{-6}$	$7x10^{-7}$	$5x10^{-7}$	$2x10^{-7}$	$2x10^{-8}$	End Melt Rel.
1.0	$8x10^{-2}$	$3x10^{-4}$	$7x10^{-3}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$4x10^{-2}$	$2x10^{-5}$	Steam Expl.
2.5	$1x10^{-1}$	$4x10^{-4}$	$7x10^{-3}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$4x10^{-2}$	$3x10^{-5}$	
4.0	$1x10^{-1}$	$5x10^{-4}$	$7x10^{-3}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$4x10^{-2}$	$3x10^{-5}$	End Vap. Rel.
16.0	$2x10^{-1}$	$9x10^{-4}$	$7x10^{-3}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$4x10^{-2}$	$3x10^{-5}$	
720	1.0	$4x10^{-3}$	$9x10^{-3}$	$6x10^{-3}$	$2x10^{-2}$	$8x10^{-4}$	$4x10^{-2}$	$3x10^{-5}$	
					Case ADβ				
0.2	2.10^{-3}	0	$5x10^{-4}$	$1x10^{-3}$	$9x10^{-6}$	$1x10^{-7}$	0	0	Just Before Melt
1.0	$6x10^{-2}$	$2x10^{-4}$	$3x10^{-3}$	$7x10^{-3}$	$1x10^{-3}$	$7x10^{-4}$	$2x10^{-4}$	$2x10^{-5}$	End Melt Rel.
2.5	$1x10^{-1}$	$5x10^{-4}$	$3x10^{-3}$	$7x10^{-3}$	$3x10^{-3}$	$8x10^{-4}$	$4x10^{-4}$	$5x10^{-5}$	
4.0	$1x10^{-1}$	$6x10^{-4}$	$4x10^{-3}$	$8x10^{-3}$	$4x10^{-3}$	$8x10^{-4}$	$4x10^{-4}$	$6x10^{-5}$	End Vap. Rel.
16.0	$3x10^{-1}$	$1x10^{-3}$	$4x10^{-3}$	$8x10^{-3}$	$5x10^{-3}$	$8x10^{-4}$	$4x10^{-4}$	$6x10^{-5}$	
720	1.0	$4x10^{-3}$	$6x10^{-3}$	$8x10^{-3}$	$5x10^{-3}$	$8x10^{-4}$	$4x10^{-4}$	$6x10^{-5}$	

TABLE 6 (CONTINUED)

TABLE 6 (CONTINUED)

Time, hr	Cumulative Fractions of Core Inventory Released to the Atmosphere[a]								Event[b]
	Xe-Kr	Org-I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La	
					Case ADFβ				
0.25	$2x10^{-3}$	0	$8x10^{-4}$	$8x10^{-3}$	$1x10^{-5}$	$2x10^{-7}$	0	0	Just Before Melt
1.0	$7x10^{-2}$	$3x10^{-4}$	$5x10^{-3}$	$1x10^{-2}$	$3x10^{-3}$	$2x10^{-3}$	$5x10^{-4}$	$5x10^{-5}$	End Melt Rel.
4.0	$6x10^{-1}$	$2x10^{-3}$	$3x10^{-2}$	$6x10^{-2}$	$2x10^{-2}$	$5x10^{-3}$	$1x10^{-2}$	$2x10^{-3}$	End Vap. Rel.
26.5	$8x10^{-1}$	$3x10^{-3}$	$4x10^{-2}$	$7x10^{-2}$	$2x10^{-1}$	$6x10^{-3}$	$1x10^{-2}$	$3x10^{-3}$	
720	1.0	$4x10^{-3}$	$5x10^{-2}$	$7x10^{-2}$	$2x10^{-1}$	$6x10^{-3}$	$1x10^{-2}$	$3x10^{-3}$	
					Case ADFε				
0.25	$2x10^{-6}$	0	$7x10^{-7}$	$2x10^{-6}$	$1x10^{-8}$	$2x10^{-10}$	0	0	Just Before Melt
1.0	$5x10^{-5}$	$2x10^{-7}$	$5x10^{-6}$	$1x10^{-5}$	$2x10^{-6}$	$1x10^{-6}$	$3x10^{-7}$	$3x10^{-8}$	End Melt Rel.
2.0	$1x10^{-4}$	$6x10^{-7}$	$6x10^{-6}$	$1x10^{-5}$	$3x10^{-6}$	$2x10^{-6}$	$5x10^{-7}$	$5x10^{-8}$	
4.0	$9x10^{-4}$	$4x10^{-6}$	$5x10^{-5}$	$8x10^{-5}$	$2x10^{-4}$	$7x10^{-6}$	$1x10^{-5}$	$3x10^{-6}$	End Vap. Rel.
18.3	$7x10^{-3}$	$3x10^{-5}$	$1x10^{-4}$	$2x10^{-4}$	$8x10^{-4}$	$2x10^{-5}$	$5x10^{-5}$	$9x10^{-6}$	
20.0	$3x10^{-1}$	$1x10^{-3}$	$1x10^{-4}$	$2x10^{-4}$	$8x10^{-4}$	$2x10^{-5}$	$5x10^{-5}$	$1x10^{-5}$	Cont. Meltthrough
720	$5x10^{-1}$	$2x10^{-3}$	$1x10^{-3}$	$2x10^{-4}$	$9x10^{-4}$	$2x10^{-5}$	$5x10^{-5}$	$1x10^{-5}$	
					Case ACDβ				
0.24	$2x10^{-3}$	0	$2x10^{-3}$	$2x10^{-3}$	$1x10^{-5}$	$2x10^{-7}$	0	0	Just Before Melt
1.0	$1x10^{-1}$	$5x10^{-4}$	$6x10^{-2}$	$2x10^{-2}$	$5x10^{-3}$	$3x10^{-3}$	$9x10^{-4}$	$9x10^{-5}$	End Melt Rel.
4.0	$3x10^{-1}$	$1x10^{-3}$	$9x10^{-2}$	$4x10^{-2}$	$1x10^{-2}$	$5x10^{-3}$	$2x10^{-3}$	$2x10^{-4}$	End Vap. Rel.
26.5	$6x10^{-1}$	$2x10^{-3}$	$9x10^{-2}$	$4x10^{-2}$	$1x10^{-2}$	$5x10^{-3}$	$2x10^{-3}$	$2x10^{-4}$	
720	1.0	$4x10^{-3}$	$9x10^{-2}$	$4x10^{-2}$	$1x10^{-2}$	$5x10^{-3}$	$2x10^{-3}$	$2x10^{-4}$	
					Case ACDε				
0.25	$2x10^{-6}$	0	$2x10^{-6}$	$2x10^{-6}$	$1x10^{-8}$	$2x10^{-10}$	0	0	Just Before Melt
1.0	$1x10^{-4}$	$5x10^{-7}$	$7x10^{-5}$	$3x10^{-5}$	$5x10^{-6}$	$3x10^{-6}$	$1x10^{-6}$	$1x10^{-7}$	End Melt Rel.
4.0	$1x10^{-3}$	$4x10^{-6}$	$1x10^{-4}$	$7x10^{-5}$	$5x10^{-5}$	$9x10^{-6}$	$4x10^{-6}$	$7x10^{-7}$	End Vap. Rel.
26.5	$8x10^{-3}$	$4x10^{-5}$	$1x10^{-4}$	$7x10^{-5}$	$6x10^{-5}$	$9x10^{-6}$	$5x10^{-6}$	$8x10^{-7}$	
720	$2x10^{-1}$	$9x10^{-4}$	$5x10^{-4}$	$7x10^{-5}$	$6x10^{-5}$	$9x10^{-6}$	$5x10^{-6}$	$8x10^{-7}$	
					Case ACDGIα				
0.26	$2x10^{-6}$	0	$2x10^{-6}$	$2x10^{-6}$	$1x10^{-8}$	$2x10^{-10}$	0	0	Just Before Melt
1.0	$1x10^{-4}$	$5x10^{-7}$	$6x10^{-5}$	$3x10^{-5}$	$5x10^{-6}$	$3x10^{-6}$	$1x10^{-6}$	$1x10^{-7}$	End Melt Rel.
1.0	$5x10^{-1}$	$2x10^{-3}$	$2x10^{-1}$	$9x10^{-2}$	$2x10^{-1}$	$1x10^{-2}$	$3x10^{-1}$	$4x10^{-4}$	Steam Expl.
4.0	$8x10^{-1}$	$3x10^{-3}$	$2x10^{-1}$	$1x10^{-1}$	$2x10^{-1}$	$1x10^{-2}$	$3x10^{-1}$	$8x10^{-4}$	End Vap. Rel.
720	$9x10^{-1}$	$4x10^{-3}$	$2x10^{-1}$	$1x10^{-1}$	$2x10^{-1}$	$1x10^{-2}$	$3x10^{-1}$	$8x10^{-4}$	

TABLE 6 (CONTINUED)

Time, hr	Cumulative Fractions of Core Inventory Released to the Atmosphere[a]								Event[b]
	Xe-Kr	Org-I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La	
					Case ACDGIδ				
0.26	$2x10^{-6}$	0	$2x10^{-6}$	$2x10^{-6}$	$1x10^{-8}$	$2x10^{-10}$	0	0	Just Before Melt
1.0	$1x10^{-4}$	$5x10^{-7}$	$7x10^{-5}$	$3x10^{-5}$	$5x10^{-6}$	$3x10^{-6}$	$1x10^{-6}$	$1x10^{-7}$	End Melt Rel.
4.0	$1x10^{-3}$	$4x10^{-6}$	$1x10^{-4}$	$7x10^{-5}$	$5x10^{-5}$	$9x10^{-6}$	$5x10^{-6}$	$7x10^{-7}$	End Vap. Rel.
17.6	$6x10^{-3}$	$3x10^{-5}$	$1x10^{-4}$	$7x10^{-5}$	$6x10^{-5}$	$9x10^{-6}$	$5x10^{-6}$	$8x10^{-7}$	
20.0	$8x10^{-1}$	$3x10^{-3}$	$1x10^{-3}$	$7x10^{-5}$	$6x10^{-5}$	$9x10^{-6}$	$5x10^{-6}$	$8x10^{-7}$	Overpressure
720	1.0	$4x10^{-3}$	$2x10^{-3}$	$7x10^{-5}$	$6x10^{-5}$	$9x10^{-6}$	$5x10^{-6}$	$8x10^{-7}$	
					Case ACDGIβ				
0.26	$2x10^{-3}$	0	$2x10^{-3}$	$2x10^{-3}$	$1x10^{-5}$	$2x10^{-7}$	0	0	Just Before Melt
1.0	$1x10^{-1}$	$5x10^{-4}$	$5x10^{-2}$	$2x10^{-2}$	$4x10^{-3}$	$3x10^{-3}$	$9x10^{-4}$	$9x10^{-5}$	End Melt Rel.
4.0	$5x10^{-1}$	$2x10^{-3}$	$8x10^{-2}$	$4x10^{-2}$	$3x10^{-2}$	$5x10^{-3}$	$3x10^{-3}$	$3x10^{-4}$	End Vap. Rel.
26.5	1.0	$4x10^{-3}$	$8x10^{-2}$	$4x10^{-2}$	$3x10^{-2}$	$5x10^{-3}$	$3x10^{-3}$	$4x10^{-4}$	
720	1.0	$4x10^{-3}$	$8x10^{-2}$	$4x10^{-2}$	$3x10^{-2}$	$5x10^{-3}$	$3x10^{-3}$	$4x10^{-4}$	
					Case ACDGIε				
0.26	$2x10^{-6}$	0	$2x10^{-6}$	$2x10^{-6}$	$1x10^{-8}$	$2x10^{-10}$	0	0	Just Before Melt
1.0	$1x10^{-4}$	$5x10^{-7}$	$7x10^{-5}$	$3x10^{-5}$	$5x10^{-6}$	$3x10^{-6}$	$1x10^{-6}$	$1x10^{-7}$	End Melt Rel.
4.0	$1x10^{-3}$	$4x10^{-6}$	$1x10^{-4}$	$7x10^{-5}$	$5x10^{-5}$	$9x10^{-6}$	$5x10^{-6}$	$7x10^{-7}$	End Vap. Rel.
17.6	$6x10^{-3}$	$2x10^{-5}$	$1x10^{-4}$	$7x10^{-5}$	$6x10^{-5}$	$9x10^{-6}$	$5x10^{-6}$	$8x10^{-7}$	
20.0	$6x10^{-1}$	$2x10^{-3}$	$1x10^{-4}$	$7x10^{-5}$	$6x10^{-5}$	$9x10^{-6}$	$5x10^{-6}$	$8x10^{-7}$	Cont. Meltthrough
720	$7x10^{-1}$	$3x10^{-3}$	$3x10^{-4}$	$7x10^{-5}$	$6x10^{-5}$	$9x10^{-6}$	$5x10^{-6}$	$8x10^{-7}$	
					Case ABα				
0.27	$2x10^{-6}$	0	$3x10^{-6}$	$9x10^{-6}$	$7x10^{-8}$	$1x10^{-9}$	0	0	Just Before Melt
1.0	$1x10^{-4}$	$9x10^{-7}$	$1x10^{-4}$	$1x10^{-4}$	$2x10^{-5}$	$1x10^{-5}$	$3x10^{-6}$	$3x10^{-7}$	End Melt Rel.
1.0	$8x10^{-1}$	$6x10^{-3}$	$6x10^{-1}$	$4x10^{-1}$	$3x10^{-1}$	$5x10^{-2}$	$4x10^{-1}$	$2x10^{-3}$	Steam Expl.
2.5	$8x10^{-1}$	$6x10^{-3}$	$6x10^{-1}$	$4x10^{-1}$	$3x10^{-1}$	$5x10^{-2}$	$4x10^{-1}$	$2x10^{-3}$	1/2 Vap. Rel.
8.3	$8x10^{-1}$	$6x10^{-3}$	$6x10^{-1}$	$4x10^{-1}$	$3x10^{-1}$	$5x10^{-2}$	$4x10^{-1}$	$2x10^{-3}$	
17.3	$9x10^{-1}$	$6x10^{-3}$	$6x10^{-1}$	$4x10^{-1}$	$3x10^{-1}$	$5x10^{-2}$	$4x10^{-1}$	$2x10^{-3}$	
720	1.0	$7x10^{-3}$	$7x10^{-1}$	$4x10^{-1}$	$3x10^{-1}$	$5x10^{-2}$	$4x10^{-1}$	$2x10^{-3}$	
					Case ABγ				
0.27	$2x10^{-6}$	0	$3x10^{-6}$	$9x10^{-6}$	$7x10^{-8}$	$1x10^{-9}$	0	0	Just Before Melt
1.0	$1x10^{-4}$	$9x10^{-7}$	$1x10^{-4}$	$1x10^{-4}$	$2x10^{-5}$	$1x10^{-5}$	$3x10^{-6}$	$3x10^{-7}$	End Melt Rel.
1.0	$8x10^{-1}$	$6x10^{-3}$	$6x10^{-1}$	$4x10^{-1}$	$8x10^{-2}$	$6x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	Overpressure
2.5	$9x10^{-1}$	$7x10^{-3}$	$7x10^{-1}$	$5x10^{-1}$	$1x10^{-1}$	$6x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	1/2 Vap. Rel.
8.3	$9x10^{-1}$	$7x10^{-3}$	$7x10^{-1}$	$5x10^{-1}$	$1x10^{-1}$	$6x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	
720	$9x10^{-1}$	$7x10^{-3}$	$7x10^{-1}$	$5x10^{-1}$	$1x10^{-1}$	$6x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	

TABLE 6 (CONTINUED)

TABLE 6 (CONTINUED)

Time, hr	Cumulative Fractions of Core Inventory Released to the Atmosphere[a]								Event[b]
	Xe-Kr	Org-I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La	
					Case ABε				
0.27	$2x10^{-6}$	0	$3x10^{-6}$	$9x10^{-6}$	$7x10^{-8}$	$1x10^{-9}$	0	0	Just Before Melt
1.0	$2x10^{-4}$	$1x10^{-6}$	$1x10^{-4}$	$1x10^{-4}$	$2x10^{-5}$	$1x10^{-6}$	$3x10^{-6}$	$3x10^{-7}$	End Melt Rel.
2.5	$7x10^{-4}$	$5x10^{-6}$	$4x10^{-4}$	$3x10^{-4}$	$9x10^{-5}$	$3x10^{-5}$	$1x10^{-5}$	$2x10^{-6}$	1/2 Vap. Rel.
5.1	$2x10^{-3}$	$1x10^{-5}$	$7x10^{-4}$	$5x10^{-4}$	$4x10^{-4}$	$5x10^{-5}$	$3x10^{-5}$	$5x10^{-6}$	
17.1	$7x10^{-3}$	$5x10^{-5}$	$7x10^{-4}$	$7x10^{-4}$	$8x10^{-4}$	$8x10^{-5}$	$6x10^{-5}$	$1x10^{-5}$	
20.0	$3x10^{-1}$	$2x10^{-3}$	$7x10^{-4}$	$8x10^{-4}$	$9x10^{-4}$	$8x10^{-5}$	$7x10^{-5}$	$1x10^{-5}$	Cont. Meltthrough
720	$4x10^{-1}$	$3x10^{-3}$	$3x10^{-3}$	$8x10^{-4}$	$1x10^{-3}$	$9x10^{-5}$	$7x10^{-5}$	$1x10^{-5}$	
					Case $S_2C\alpha$				
4.7	$8x10^{-3}$	0	$4x10^{-3}$	$9x10^{-3}$	$2x10^{-5}$	$2x10^{-7}$	0	0	Just Before Melt
6.0	$2x10^{-1}$	$1x10^{-3}$	$1x10^{-1}$	$1x10^{-1}$	$2x10^{-2}$	$1x10^{-2}$	$4x10^{-3}$	$4x10^{-4}$	End Melt Rel.
6.0	$4x10^{-1}$	$3x10^{-3}$	$3x10^{-1}$	$2x10^{-1}$	$1x10^{-1}$	$2x10^{-2}$	$1x10^{-1}$	$7x10^{-4}$	Steam Expl.
7.8	$5x10^{-1}$	$4x10^{-3}$	$3x10^{-1}$	$2x10^{-1}$	$1x10^{-1}$	$3x10^{-2}$	$1x10^{-1}$	$8x10^{-4}$	1/2 Vap. Rel.
9.3	$5x10^{-1}$	$5x10^{-3}$	$3x10^{-1}$	$2x10^{-1}$	$1x10^{-1}$	$3x10^{-2}$	$1x10^{-1}$	$9x10^{-4}$	End Vap. Rel.
24	$7x10^{-1}$	$6x10^{-3}$	$3x10^{-1}$	$2x10^{-1}$	$1x10^{-1}$	$3x10^{-2}$	$1x10^{-1}$	$1x10^{-3}$	
720	1.0	$8x10^{-3}$	$3x10^{-1}$	$2x10^{-1}$	$1x10^{-1}$	$3x10^{-2}$	$1x10^{-1}$	$1x10^{-3}$	
					Case $S_2C\delta$				
4.7	$8x10^{-3}$	0	$4x10^{-3}$	$9x10^{-3}$	$2x10^{-5}$	$2x10^{-7}$	0	0	Just Before Melt
6.0	$2x10^{-1}$	$1x10^{-3}$	$1x10^{-1}$	$1x10^{-1}$	$2x10^{-2}$	$1x10^{-2}$	$4x10^{-3}$	$4x10^{-3}$	End Melt Rel.
7.8	$3x10^{-1}$	$2x10^{-3}$	$2x10^{-1}$	$1x10^{-1}$	$5x10^{-2}$	$1x10^{-2}$	$6x10^{-3}$	$8x10^{-4}$	1/2 Vap. Rel.
9.3	$4x10^{-1}$	$3x10^{-3}$	$2x10^{-1}$	$1x10^{-1}$	$8x10^{-2}$	$2x10^{-2}$	$8x10^{-3}$	$1x10^{-3}$	End Vap. Rel.
24	$8x10^{-1}$	$6x10^{-3}$	$2x10^{-1}$	$2x10^{-1}$	$1x10^{-1}$	$2x10^{-2}$	$1x10^{-2}$	$2x10^{-3}$	
720	1.0	$7x10^{-3}$	$2x10^{-1}$	$2x10^{-1}$	$1x10^{-1}$	$2x10^{-2}$	$1x10^{-2}$	$2x10^{-3}$	
					Case TMLB'γ				
2.6	$7x10^{-4}$	0	$4x10^{-4}$	$8x10^{-4}$	$2x10^{-6}$	$2x10^{-8}$	0	0	Just Before Melt
3.6	$2x10^{-2}$	$1x10^{-4}$	$1x10^{-2}$	$1x10^{-2}$	$2x10^{-3}$	$1x10^{-3}$	$3x10^{-4}$	$3x10^{-5}$	End Melt Rel.
3.7	$8x10^{-1}$	$6x10^{-3}$	$6x10^{-1}$	$4x10^{-1}$	$8x10^{-2}$	$5x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	Overpressure
6.7	$8x10^{-1}$	$6x10^{-3}$	$7x10^{-1}$	$4x10^{-1}$	$1x10^{-1}$	$6x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	End Vap. Rel.
17	$9x10^{-1}$	$7x10^{-3}$	$7x10^{-1}$	$5x10^{-1}$	$2x10^{-1}$	$6x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	
720	1.0	$7x10^{-3}$	$7x10^{-1}$	$5x10^{-1}$	$2x10^{-1}$	$6x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	
					Case TMLB'δ				
2.6	$7x10^{-4}$	0	$4x10^{-4}$	$8x10^{-4}$	$2x10^{-6}$	$2x10^{-8}$	0	0	Just Before Melt
3.6	$2x10^{-2}$	$1x10^{-4}$	$1x10^{-2}$	$1x10^{-2}$	$2x10^{-3}$	$1x10^{-3}$	$3x10^{-4}$	$3x10^{-5}$	End Melt Rel.
3.7	$8x10^{-1}$	$6x10^{-3}$	$6x10^{-1}$	$4x10^{-1}$	$8x10^{-2}$	$5x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	Overpressure
6.7	$8x10^{-1}$	$6x10^{-3}$	$6x10^{-1}$	$4x10^{-1}$	$1x10^{-1}$	$5x20^{-2}$	$2x10^{-2}$	$2x10^{-3}$	End Vap. Rel.
17	$9x10^{-1}$	$7x10^{-3}$	$6x10^{-1}$	$4x10^{-1}$	$2x10^{-1}$	$5x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	
720	1.0	$7x10^{-3}$	$7x10^{-1}$	$4x10^{-1}$	$2x10^{-1}$	$5x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	

TABLE 6 (CONTINUED)

Time, hr	Cumulative Fractions of Core Inventory Released to the Atmosphere[a]								Event[b]
	Xe-Kr	Org-I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La	
					Case TMLB'α				
2.6	$7x10^{-4}$	0	$4x10^{-4}$	$8x10^{-4}$	$2x10^{-6}$	$2x10^{-8}$	0	0	Just Before Melt
3.6	$2x10^{-2}$	$1x10^{-4}$	$1x10^{-2}$	$1x10^{-2}$	$2x10^{-3}$	$1x10^{-3}$	$3x10^{-4}$	$3x10^{-5}$	End Melt Rel.
3.7	$8x10^{-1}$	$6x10^{-3}$	$7x10^{-1}$	$4x10^{-1}$	$3x10^{-1}$	$5x10^{-2}$	$4x10^{-1}$	$2x10^{-3}$	Steam Expl.
6.7	$8x10^{-1}$	$6x10^{-3}$	$7x10^{-1}$	$4x10^{-1}$	$3x10^{-1}$	$5x10^{-2}$	$4x10^{-1}$	$2x10^{-3}$	End Vap. Rel.
17	$9x10^{-1}$	$6x10^{-3}$	$7x10^{-1}$	$4x10^{-1}$	$3x10^{-1}$	$5x10^{-2}$	$4x10^{-1}$	$2x10^{-3}$	
720	1.0	$7x10^{-3}$	$7x10^{-1}$	$4x10^{-1}$	$3x10^{-1}$	$5x10^{-2}$	$4x10^{-1}$	$2x10^{-3}$	

(a) Te includes Se And Sb
Ru includes Mo, Pd, Rh, and Tc
La includes Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, and Nb.

(b) Notations used are defined as follows:
Just Before Melt - Just before core melting begins
1/2 Melt Rel. - Melt release is one-half complete
End Melt Rel. - Melt release is complete
1/2 Val. Rel. - Vaporization release is one-half complete
End Vap. Rel. - Vaporization release is complete
PV Meltthrough - Pressure vessel maltthrough
Cont. Meltthrough- Containment failure by meltthrough
Steam Expl. - Containment failure by steam explosion
Overpressure - Containment failure by overpressure
Just Before OP - Just before containment failure by overpressure
End Leak - Containment leakage complete

TABLE 7 BWR LARGE LOCA EVENT TREE NOTATION

Letter	Symbol	Meaning
A	PB	Pipe Break
B	EP	Electric Power
C	Scram	Control Rod Insertion
D	VS	Vapor Suppression
E	ECO	Emergency Core Cooling Operation
F	ECF	Emergency Core Cooling Function
G	CL	Containment Leakage greater than 100 percent per day
H	CSRS	Core Spray Recirculation System
I	LPCRS	Low Pressure Core Recirculation System
J	HPSW	High Pressure Service Water
α	VSE	Containment failure due to a steam explosion in the reactor vessel
β	CSE	Containment failure due to a steam explosion in containment
γ	OP	Containment failure by overpressure
δ	DWL	Containment isolation failure in the drywell
ε	WWL	Containment isolation failure in the wetwell
ζ	LCL	Containment leakage greater than 2400 percent per day
η	SCF	Secondary Containment Failure
θ	SGTS	Standby Gas Treatment System failure

TABLE 8 BWR ACCIDENT SUMMARY

Sequence		Core Melting			Reactor Vessel Meltthrough (c)		Containment Overpressure	Containment Meltthrough		
		Start, min	End,[a] min	Pressure, psia	Time, min	Pressure, psia	Failure, min	Start,[b] min	Pressure, psia	End, min
1.	A	—	—	—	—	—	—	—	—	—
2.	AJ	1520	1640	15	1730	15	1500	1750	15	5000
3.	AI	1520	1640	15	1730	15	1500	1750	15	5000
4.	AH	—	—	—	—	—	—	—	—	—
5.	AHJ	1520	1640	15	1730	15	1500	1750	15	5000
6.	AHI	20	80	52	140	128	220	160	165	2000
7.	AG	—	—	—	—	—	—	—	—	—
8.	AGJ	270	330	15	390	15	—	410	15	2500
9.	AGI	270	330	15	390	15	—	410	15	2500
10.	AGH	—	—	—	—	—	—	—	—	—
11.	AHGJ	270	30	15	390	15	—	410	15	2500
12.	AGHI	20	80	24	140	19	—	160	34	2000
13.	AF	5	150	58	210	58	290	230	165	2000
14.	AFG	5	150	18	210	16	—	230	34	2000
15a.	AE	20	150	17	210	82	640	230	107	2000
15b.	AE	20	150	17	180	17	640	200	107	2000
16a.	AEG	20	150	15	210	47	—	230	66	2000
16b.	AEG	20	150	15	180	15	—	200	66	2000
17.	AD	—	—	—	—	—	—	—	—	—
18.	ADJ	420	510	15	600	15	0.5	620	15	3000
19.	ADI	420	510	15	600	15	0.5	620	15	3000
20.	ADH	—	—	—	—	—	—	—	—	—
21.	ADHJ	420	510	15	600	15	0.5	620	15	3000
22.	ADHI	20	80	15	140	15	0.5	160	15	2000
23.	ADF	5	150	15	210	15	0.5	230	15	2000
24a.	ADE	20	150	15	210	15	0.5	230	15	2000
24b.	ADE	20	150	15	180	15	0.5	200	15	2000
25.	AC	5	150	58	210	58	290	230	165	2000
26.	ACG	5	150	18	210	16	—	230	34	2000
27.	ACD	5	150	15	210	15	0.5	230	15	2000
28.	AB	20	150	17	180	17	640	200	107	2000
29.	ABG	20	150	15	180	15	—	200	66	2000
30.	ABD	20	150	15	180	15	0.5	200	15	2000
31.	TC	30	180	93	180	93	190	200	15	2000
32.	TW	1740	1860	15	1950	15	1540	1970	15	5000
33.	TQUV	115	180	56	210	57	300	230	162	2000

(a) End of core melting is taken as ~80 percent molten.

(b) After the initial rapid interaction between the molten core and concrete.

(c) For accident sequences in which the primary system is at higher pressure during core melting the reactor vessel fails by a combination of melting and pressure stress.

TABLE 9 BWR CONTAINMENT EVENT TREE PROBABILITIES

Sequence		Containment Failure Mode Probabilities										
		α	β	γ	$\varepsilon\zeta$	$\varepsilon\eta$	$\varepsilon\theta$	ε	$\delta\zeta$	$\delta\eta$	$\delta\theta$	δ
1.	A	--	--	--	--	--	--	--	--	--	--	--
2.	AJ	0.01	0	1	0	0	0	0	0	0	0	0
3.	AI	0.01	0	1	0	0	0	0	0	0	0	0
4.	AH	--	--	--	--	--	--	--	--	--	--	--
5.	AHJ	0.01	0	1	0	0	0	0	0	0	0	0
6.	AHI	0.01	0.09	0.90	0	0	0	0	0	0	0	0
7.	AG	--	--	--	--	--	--	--	--	--	--	--
8.	AGJ	0.01	0.05	0	0.02	0.00	0.00	0.02	0.04	0.00	0.00	0.86
9.	AGI	0.01	0.05	0	0.02	0.00	0.00	0.02	0.04	0.00	0.00	0.86
10.	AGH	--	--	--	--	--	--	--	--	--	--	--
11.	AGHJ	0.01	0.05	0	0.02	0.00	0.00	0.02	0.04	0.00	0.00	0.86
12.	AGHI	0.01	0.00	0	0.02	0.00	0.00	0.02	0.05	0.00	0.00	0.90
13.	AF	0.04	0.18	0.78	0	0	0	0	0	0	0	0
14.	AFG	0.04	0.01	0	0.02	0.00	0.00	0.02	0.04	0.00	0.00	0.86
15a.	AE	0.01	0.07	0.92	0	0	0	0	0	0	0	0
15b.	AE	0	0.00	1	0	0	0	0	0	0	0	0
16a.	AEG	0.01	0.05	0	0.02	0.00	0.00	0.02	0.04	0.00	0.00	0.86
16b.	AEG	0	0.00	0	0.02	0.00	0.00	0.02	0.05	0.00	0.00	0.91
17.	AD	--	--	--	--	--	--	--	--	--	--	--
18.	ADJ	0.01	0	1	0	0	0	0	0	0	0	0
19.	ADI	0.01	0	1	0	0	0	0	0	0	0	0
20.	ADH	--	--	--	--	--	--	--	--	--	--	--
21.	ADHJ	0.01	0	1	0	0	0	0	0	0	0	0
22.	ADHI	0.01	0	1	0	0	0	0	0	0	0	0
23.	ADF	0.01	0	1	0	0	0	0	0	0	0	0
24a.	ADE	0.01	0	1	0	0	0	0	0	0	0	0
24b.	ADE	0	0	1	0	0	0	0	0	0	0	0
25.	AC	0.04	0.18	0.78	0	0	0	0	0	0	0	0
26.	ACG	0.04	0.01	0	0.02	0.00	0.00	0.02	0.04	0.00	0.00	0.86
27.	ACD	0	0	1	0	0	0	0	0	0	0	0
28.	AB	0	0.00	1.00	0	0	0	0	0	0	0	0
29.	ABG	0	0.00	0	0.02	0.00	0.00	0.02	0.05	0.00	0.00	0.91
30.	ABD	0	0	1	0	0	0	0	0	0	0	0
31.	TC	0.01	0	0.99	0	0	0	0	0	0	0	0
32.	TW	0.01	0	1	0	0	0	0	0	0	0	0
33.	TQUV	0.01	0	0.99	0	0	0	0	0	0	0	0

TABLE 10. BWR TIME DEPENDENT LEAK RATES

Subsequence	Puff Release		Leakage					Flow to Suppression Pool, v/o/hr[a,b]	Remarks
	Time, min	Fraction, v/o[a]	Time Interval, min	Leak Rate, v/o/hr[a]	Pressure, psia	Temperature, F Drywell	Temperature, F Wetwell		
AJγ	--	--	0-1500	0.021	175	360	360	--	Heating of suppression pool water
			-1520	550	15	212	212	--	Containment failure and core heatup
			-1640	550	15	212	212	--	Boiloff and core melting
			-1730	230	15	212	212	--	Boiloff and Reactor vessel melting
			-1750	840	15	212	212	--	Boiloff and concrete decomposition
			-2000	260	15	212	212	--	Boiloff and concrete decomposition
			2000-	28	15	212	212	--	Concrete decomposition
AGJδ	--	--	0-0.5	83	43	272	126	--	Blowdown
			-8	42	22	233	141	--	Equalization of noncondensables
			-250	--	15	190	190	--	Pump cavitation
			-270	83	34	258	195	250	Boiloff and core heating
			-330	83	76	308	205	250	Boiloff and core melting
			-340	83	102	329	211	250	Reactor vessel boiloff completed
			-390	83	64	297	211	0	Reactor vessel melting
			-410	83	92	322	211	236	Boiloff and concrete decomposition
			-630	83	100	328	216	14	Boiloff and concrete decomposition
			-790	83	16	216	216	0	Concrete decomposition
			790-	62	16	216	216	0	Concrete decomposition
AGJε	--	--	0-0.5	83	43	272	126	--	Blowdown
			-3	83	46	267	132	--	Safety system operation
			-5	--	16	136	136	--	Equalization of noncondensables
			-250	2.5	15	190	190	--	Pump cavitation
			-270	7	16	225	196	1476	Boiloff and core heating
			-330	83	61	297	214	740	Boiloff and core melting
			-340	83	55	290	217	426	Reactor vessel boiloff complete
			-390	83	33	256	217	--	Reactor vessel melting
			-410	83	50	285	221	407	Boiloff and concrete decomposition
			-440	83	39	270	224	192	Boiloff and concrete decomposition
			-630	83	31	258	240	242	Boiloff and concrete decomposition
			630-	41	25	246	240	41	Concrete decomposition
ADJε	--	--	0-0.5	7338	15	222	130	--	Blowdown
			-5	--	15	222	140	--	Equialization of noncondesables
			-250	2.5	15	222	190	--	Pump cavitation
			-270	237	15	222	196	1476	Boiloff and core heating
			-312	237	15	222	212	1476	Boiloff and core melting
			-330	1750	15	222	212	1476	Boiloff and core melting
			-340	1530	15	222	212	1276	Reactor vessel boiloff completed
			-390	--	15	212	212	--	Reactor vessel melting
			-410	890	15	222	212	751	Boiloff and concrete decomposition
			-630	535	15	222	212	451	Boiloff and concrete decomposition
			630-	67	15	222	212	56	Concrete decomposition

TABLE 10. (CONTINUED)

Subsequence	Puff Release Time, min.	Puff Release Fraction, v/o[a]	Leakage Time Interval, min.	Leakage Leak Rate, v/o/hr[a]	Leakage Pressure, psia	Leakage Temperature, F Drywell	Leakage Temperature, F Wetwell	Flow to Suppression Pool, v/o/hr[a,b]	Remarks
AFα	20	59	0-5	0.021	24	134	134	--	Core heating
			-20	0.021	36	139	139	58	Initial core melting
			-150	585	15	212	139	--	Core melting and boiloff
			-270	190	15	212	139	--	Boiloff and reactor vessel melting
			-290	800	15	212	139	--	Boiloff and concrete decomposition
			-700	210	15	212	139	--	Boiloff and concrete decomposition
			700-	18	15	212	139	--	Concrete decomposition
AFβ	210	91	0-5	0.021	24	134	134	--	Core heating
			-20	0.021	36	139	139	58	Initial core melting
			-150	0.021	58	158	158	4.2	Core melting
			-210	0.021	58	161	161	--	Reactor vessel melting
			-230	210	15	162	162	--	CO_2 leakage only
			-290	15		165	165	--	CO_2 leakage only
			290-	12		166	166	--	CO_2 leakage only
AFγ	290	91	0-5	0.021	24	134	134	--	Core heating
			-20	0.021	36	139	139	58	Initial core melting
			-150	0.021	58	158	158	4.2	Core melting
			-210	0.021	58	161	161	0	Reactor vessel melting
			-230	0.021	165	162	162	36	Concrete decomposition and CO_2 leakage
			-290	0.021	175	165	165	2.5	Concrete decomposition and CO_2 leakage
			-350	15	15			--	Concrete decomposition and CO_2 leakage
			350-	12	15			--	Concrete decomposition and CO_2 leakage
AFGε	--	--	0-0.5	83	41	272	126	--	Blowdown
			-3	83	37	267	131	--	Safety system operation
			-5	23	18	134	134	--	Equalization of noncondensables
			-20	52	27	139	139	215	Initial core melting
			-150	49	18	158	158	43	Core melting
			-210	21	16	161	161	14	Reactor vessel melting
			-230	83	34	263	162	780	Concrete decomposition
			-240	83	27	250	163	12	Concrete decomposition
			-265	42	21	238	165	12	Concrete decomposition
			265-	20	17	228	170	18	Concrete decomposition
AFGεζ	--	--	0-0.5	7338	15	222	126	--	Blowdown
			-20	321	15	222	139	321	Initial core melting
			-150	50	15	222	158	50	Core melting
			-210	--	15	213	161	--	Reactor vessel melting
			-230	203	15	222	162	203	Concrete decomposition
			230-	22			165	20	Concrete decomposition

TABLE 10. (CONTINUED)

Subsequence	Puff Release Time, min.	Puff Release Fraction, v/o[a]	Leakage Time Interval, min.	Leakage Leak Rate, v/o/hr[a]	Leakage Pressure, psia	Leakage Temperature, F Drywell	Leakage Temperature, F Wetwell	Flow to Suppression Pool, v/o/hr[a,b]	Remarks
AEδ (dry)	130	88	0-20	0.003	17	126	126	—	Core heating
			-150	0.003	17	126	126	0.065	Core melting
			-180	0.003	17	126	126	--	Reactor vessel melting
			-200	1030	15	212	126	--	Boiloff and concrete decomposition
			-460	420	15	212	126	--	Boiloff and concrete decomposition
			-520	48	15	212	126	--	Concrete decomposition
			520-	41	15	212	126	--	Concrete decomposition
AEγ (dry)	640	91	0-20	0.003	17	126	126	--	Core heating
			-150	0.003	17	126	126	0.065	Core melting
			-180	0.003	17	126	126	0	Reactor vessel melting
			-200	0.021	107	342	132	49	Boiloff and concrete decomposition
			-640	0.021	175	371	136	2.5	Boiloff and concrete decomposition
			-700	41	15	212	136	--	Concrete decomposition
			700-	34	15	212	136	--	Concrete decomposition
AEGδ (dry)	—	—	0-0.5	83	43	272	126	--	Blowdown
			-3.5	83	40	267	126	--	Equalization of noncondensables
			-20	--	17	126	126	--	Core heating
			-150	--	15	126	126	--	Core melting
			-180	--	15	126	126	--	Reactor vessel melting
			-200	83	66	299	130	250	Boiloff and concrete decomposition
			-420	83	79	311	137	49	Boiloff and concrete decomposition
			-600	83	16	216	137	0	Concrete decomposition
			600-	62	16	216	137	0	Concrete decomposition
ADJγ	0.5	91	0-400	51	15	212	190	—	Heating of suppression pool water and pump cavitation
			-420	51	15	212	190	—	Core heating
			-510	780	15	212	190	—	Core melting
			-600	390	15	212	190	—	Boiloff and reactor vessel melting
			-620	1390	15	212	190	—	Boiloff and concrete decomposition
			-970	350	15	212	190	—	Boiloff and concrete decomposition
			970-	30	15	212	190	--	Concrete decomposition
ADFγ	0.5	91	0-5	51	15	97	97	--	Core heating
			-150	51	15	122	122	—	Core melting
			-210	51	15	125	125	--	Reactor vessel melting
			-230	210	15	126	126	--	Concrete decomposition and CO_2 leakage
			-290	15	15	129	129	--	Concrete decomposition and CO_2 leakage
			290-	12	15	130	130	--	Concrete decomposition and CO_2 leakage

TABLE 10. (CONTINUED)

Subsequence	Puff Release Time, min.	Puff Release Fraction, v/o[a]	Leakage Time Interval, min.	Leakage Leak Rate, v/o/hr[a]	Leakage Pressure, psia	Leakage Temperature, F Drywell	Leakage Temperature, F Wetwell	Flow to Suppression Pool, v/o/hr[a,b]	Remarks
ADEγ (dry)	0.5	91	0-20	0.25	15	212	126	—	Core heating
			~150	0.25	15	212	126	--	Core melting
			~180	0.25	15	212	126	--	Reactor vessel melting
			~200	1030	15	212	126	--	Boiloff and concrete decomposition
			~460	420	15	212	126	--	Boiloff and concrete decomposition
			~520	48	15	212	126	--	Concrete decomposition
			520~	41	15	212	126	—	Concrete decomposition
TC-α			0-30	0.021	48	250	250	—	Primary system at 1100-1250 psia, boiloff through relief and safety valves, core cooled
			30-50	0.021	71	250	250	3650[c,e]	Primary system at 1100 psia, core heatup and initial melt, fission products released to suppression pool
	50	99/79[d]	50	--	15	212	212	--	Steam Explosion, Reactor Vessel, and Containment Failure
			50-180	585	15	212	212	--	Completion of Core melt
			180-240	190	15	212	212	--	Reactor Vessel Melt
			240-260	300	15	212	212	--	Initial Concrete Attack
			260-320	22	15	212	212	--	Concrete Decomposition
			320-380	17	15	212	212	--	Concrete Decomposition
TC-γ			0-30	0.021	48	250	250	--	Primary system at 1100-1250 psia, boiloff through relif and safety valves, core cooled
			30-50	0.021	71	250	250	3650[c,e]	Primary system at 1100 psia, core heatup and initial melt, fission products released to suppression pool
			50-180	0.031	93	250	250	3650[c,e]	Primary system at 1100 psia, completion of core melt, reactor vessel failure
			180-190	0.021	175	337	250	390	Initial Concrete Attack
	190	92	190	--	15	212	212	--	Containmnet Failure
			190-200	600	15	212	212	—	Concrete Decomposition
			200-260	43	15	212	212	--	Concrete Decomposition
			260-320	34	15	212	212	--	Concrete Decomposition
TW-α			0-1540	0.021	175	360	360	--	Heating of suppression pool water, core cooled
			1540	--	15	212	212	—	Containment Failure, core cooled
			1540-1740	570[e]	15	212	212		Boiloff through relief valves, core cooled
			1740-1860	660[e]	15	212	212		Boiloff and core melt
	1860	80	1860	--	15	212	212	--	Steam explosion, primary system failure
			1860-1980	115	15	212	212	--	Reactor Vessel melt
			1980-2000	300	15	212	212	--	Initial Concrete Attack
			2000-2060	14	15	212	212	--	Concrete decomposition
			2060-2120	12	15	212	212	--	Concrete decomposition
TWγ or TWγ*			0-1540	0.021	175	360	360	--	Heating of suppression pool water, core cooled
			1540	--	15	212	212	--	Containment Failure, core cooled
			1540-1740	570[e]	15	212	212		Boiloff through relief valves, core cooled
			1740-1860	660[e]	15	212	212		Boiloff and core melt
			1860-1950	660[e]	15	212	212		Reactor Vessel Melt
			1950-1970	600	15	212	212	--	Initial Concrete Attack
			1970-2030	28	15	212	212	--	Concrete Decomposition
			2030-2090	24	15	212	212	--	Concrete Decomposition

(a) Normalized to a containment free volume of 278,000 ft^3.
(b) Flow of gas and vapor, excluding that during primary system blowdown.
(c) Normalized to a primary system free volume of 28,835 ft^3.
(d) Fractions of primary cooling system/primary containment volume.
(e) Flow from primary coolant system directly to suppression pool.

TABLE 11 FISSION PRODUCT RELEASE VALUES FOR CORRAL-BWR CALCULATIONS

Sequence	Release Component[a]	Time of Release, minutes	Fraction of Core Inventory Released to Containment[b]						
			Xe	I	Cs	Te	Sr	Ru	La
A	Gap	1	.030	.0017	.0050	10^{-5}	10^{-7}	0	0
AJγ	Gap	Ignore to simplify calculations							
	Melt	1520-1640	.870	.860	.700	.150	.100	.030	.003
	Vaporiz.	1730-1850	.100	.100	.170	.850	.010	.050	.010
AFγ	Gap	10	.465	.047	.048	.0076	.005	.0015	.00015
	Melt	20-150	.435	.043	.035	.0075	.005	.0015	.00015
	Vaporiz.	21-330	.100	.100	.170	.850	.010	.050	.010
AFβ	Gap	10	.465	.047	.048	.0076	.005	.0015	.00015
	Melt	20-150	.435	.043	.035	.0075	.005	.0015	.00015
	Vaporiz.	21-330	.100	.100	.170	.850	.010	.050	.010
AFα	Gap	10	.465	.047	.048	.0076	.005	.0015	.00015
	St. Expl.	20	.0157	.0135	-	.127	-	.218	-
	Melt	20-150	.435	.430	.350	.075	.050	.015	.0015
	Vaporiz.	270-390	.050	.050	.085	.425	.005	.025	.005
AFα (wet)	Gap	1	.030	.017	.050	10^{-4}	10^{-6}	0	0
	Melt	20-150	.870	.883	.760	.150	.100	.030	.003
	St. Expl.	150	.045	.045	-	.255	-	.437	-
	Vaporiz.	21-330	.050	.050	.095	.425	.005	.025	.005
AEγ (dry)	Gap	1	.030	.040	.130	10^{-3}	10^{-5}	0	0
	Melt	85-150	.583	.575	.468	.100	.067	.020	.002
	Vaporiz.	180-300	.387	.385	.402	.899	.043	.060	.011
AEβ (dry)	Gap	1	.030	.040	.130	10^{-3}	10^{-5}	0	0
	Melt	85-150	.583	.575	.468	.100	.067	.020	.002
	St. Expl.	180	.296	.286	.230	.086	.033	.071	.001
	Vaporiz.	180-300	.050	.050	.085	.425	.005	.025	.005
ADJγ	Gap	2	.030	.040	.130	10^{-3}	10^{-5}	0	0
	Melt	420-510	.870	.860	.700	.150	.100	.030	.003
	Vaporiz.	600-720	.100	.100	.170	.849	.010	.050	.010
ADFγ	Gap	10	.465	.047	.048	.0076	.005	.0015	.00015
	Melt	20-150	.435	.043	.035	.0075	.005	.0015	.00015
	Vaporiz.	210-330	.100	.100	.170	.850	.010	.050	.010
ADEγ (dry)	Gap	2	.030	.040	.130	10^{-3}	10^{-5}	0	0
	Melt	85-150	.583	.575	.468	.100	.067	.020	.002
	Vaporiz.	180-300	.387	.385	.402	.899	.043	.060	.011
AGJδ	Gap	1	.030	.0017	.005	10^{-5}	10^{-7}	0	0
	Melt	270-330	.870	.883	.760	.150	.100	.030	.003
	Vaporiz.	370-490	.100	.100	.190	.850	.010	.050	.010

TABLE 11 (CONTINUED)

Sequence	Release Component[a]	Time of Release, minutes	Fraction of Core Inventory Released to Containment[b]						
			Xe	I	Cs	Te	Sr	Ru	La
AGJδθ	Gap	1	.030	.0017	.005	10^{-5}	10^{-7}	0	0
	Melt	270-330	.870	.883	.760	.150	.100	.030	.003
	Vaporiz.	370-490	.100	.100	.190	.850	.010	.050	.010
AEGδ (dry)	Gap	1	.030	.017	.050	10^{-4}	10^{-6}	0	0
	Melt	85-150	.580	.590	.507	.100	.067	.020	.002
	Vaporiz.	180-300	.390	.393	.443	.900	.043	.060	.011
AEGδη (dry)	Gap	1	.030	.017	.050	10^{-4}	10^{-6}	0	0
	Melt	85-150	.580	.590	.507	.100	.067	.020	.002
	Vaporiz.	180-300	.390	.393	.443	.900	.043	.060	.011
AGJε, εη, and εζ	Gap	1	.030	.0017	.0050	10^{-5}	10^{-7}	0	0
	Melt	270-330	.870	.883	.760	.150	.100	.030	.003
	Vaporiz.	370-490	.100	.100	.190	.850	.010	.050	.010
AFGε, εη, and εζ	Gap	10	.465	.047	.048	.0076	.005	.0015	.00015
	Melt	20-150	.435	.043	.035	.0075	.005	.0015	.00015
	Vaporiz.	210-330	.100	.100	.170	.850	.010	.050	.010
TCγ	Gap	-	-	-	-	-	-	-	-
	Melt	180	.900	-	-	-	-	-	-
	Vaporiz.	180-300	.100	.100	.190	.850	.010	.050	.010
TCα	Gap	-	-	-	-	-	-	-	-
	ST. Expl.	50	.510	.050	.004	.256	5×10^{-4}	.437	1.5×10^{-5}
	Melt	50-180	.435	.442	.380	.075	.050	.015	1.5×10^{-3}
	Vaporiz.	240-360	.050	.050	.095	.425	.005	.025	.005
TWγ and TWγ	Gap	-	-	-	-	-	-	-	-
	Melt	1740-1860	.900	.900	.810	.150	.100	.030	.003
	Vaporiz.	1950-2070	.100	.100	.190	.850	.010	.050	.010

[a] Gap means the gap release component.
Melt means the core melt release component.
St. Expl. means the steam explosion release component.
Vaporiz. means the vaporization release component.

[b] Xe also includes Kr.
I also includes Br.
Cs also includes Rb.
Te also includes Se and Sb.
Sr also includes Ba.
Ru also includes Mo, Pd, Rh, and Tc.
La also includes Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, and Nb.

TABLE 12 CONSTANTS AND PARAMETERS USED IN CORRAL-BWR RUNS

Reactor Compartment Data

Compartment	Wall Area, ft^2	Floor Area, ft^2	Height, ft	Volume, ft^3
Drywell	1.48×10^4	3.5×10^3	100	1.59×10^5
Wetwell	1.63×10^4	1.1×10^4	30	1.19×10^5
Annulus	-	-	100	2.78×10^1
Reactor Building	3.14×10^4	2.0×10^4	56	1.10×10^6
SGTS Filter	-	-	0.1	2.78×10^1

Fission Product Cleanup Specifications

Decontamination factor for suppression pool scrubbing
1 (noble gases and organic iodide)
100 (all other species)
SGTS filter efficiencies
99% (organic iodide)
99% (elemental iodine)
99% (particulate species)
zero (noble gases)

Other Parameters

Organic iodide conversion ratio = 0.7%[a]
Aerosol particle diameters, μm
Early 15
Late 5
Time interval of aerosol particle diameter change = 4 hr
Flow rate through SGTS filters, cfm
Minimum 2000
Maximum 10000
Decay heat load limit for SGTS filters, watts
HEPA units 5.4×10^4
Charcoal units 6.4×10^4

(a) Exception for the following cases:
ADFδ uses 0.13%
AFα uses 0.35%

TABLE 13 RESULTS OF CORRAL-BWR CALCULATIONS

Time hr	Cumulative Fractions of Core Inventory Released to the Atmosphere[a]								Event[b]
	Xe-Kr	Org-I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La	
					Case A				
1.0	6×10^{-7}	2×10^{-12}	3×10^{-12}	4×10^{-10}	8×10^{-13}	8×10^{-15}	0	0	
2.0	2×10^{-6}	7×10^{-12}	8×10^{-12}	9×10^{-10}	2×10^{-12}	2×10^{-14}	0	0	
3.5	6×10^{-6}	3×10^{-11}	2×10^{-11}	1×10^{-9}	3×10^{-12}	3×10^{-14}	0	0	
8.5	3×10^{-5}	1×10^{-10}	4×10^{-11}	3×10^{-9}	6×10^{-12}	6×10^{-14}	0	0	
38.5	3×10^{-4}	1×10^{-9}	6×10^{-11}	4×10^{-9}	8×10^{-12}	8×10^{-14}	0	0	
72	5×10^{-4}	2×10^{-9}	6×10^{-11}	4×10^{-9}	8×10^{-12}	8×10^{-14}	0	0	Gap Rel. Only
					Case AJγ (Release Through Annulus to Atmosphere)				
26.3	4×10^{-1}	3×10^{-3}	2×10^{-2}	3×10^{-2}	5×10^{-3}	4×10^{-3}	1×10^{-3}	1×10^{-4}	
27.3	8×10^{-1}	7×10^{-3}	4×10^{-2}	5×10^{-2}	1×10^{-2}	8×10^{-3}	2×10^{-3}	2×10^{-4}	End Melt Rel.
31.3	1.0	7×10^{-3}	5×10^{-2}	7×10^{-2}	7×10^{-2}	9×10^{-3}	6×10^{-3}	9×10^{-4}	After Vap. Rel.
49.3	1.0	7×10^{-3}	5×10^{-2}	7×10^{-2}	7×10^{-2}	9×10^{-3}	6×10^{-3}	9×10^{-4}	
					Case AJγ (Release Directly to Atmosphere)				
26.3	4×10^{-1}	3×10^{-3}	2×10^{-1}	8×10^{-2}	2×10^{-2}	1×10^{-2}	3×10^{-3}	3×10^{-4}	
27.3	8×10^{-1}	7×10^{-3}	4×10^{-1}	2×10^{-1}	4×10^{-2}	2×10^{-2}	7×10^{-3}	7×10^{-4}	End Melt Rel.
31.3	1.0	7×10^{-3}	5×10^{-1}	2×10^{-1}	2×10^{-1}	3×10^{-2}	2×10^{-2}	3×10^{-3}	After Vap. Rel.
49.3	1.0	7×10^{-3}	5×10^{-1}	2×10^{-1}	2×10^{-1}	3×10^{-2}	2×10^{-2}	3×10^{-3}	
					Case AFγ				
0.3	2×10^{-7}	1×10^{-12}	2×10^{-12}	2×10^{-10}	3×10^{-11}	2×10^{-11}	6×10^{-12}	6×10^{-13}	Just Before Melt
2.5	5×10^{-5}	2×10^{-9}	4×10^{-10}	1×10^{-8}	2×10^{-9}	1×10^{-9}	4×10^{-10}	4×10^{-11}	End Melt Rel.
4.0	1×10^{-4}	7×10^{-9}	1×10^{-9}	2×10^{-8}	1×10^{-8}	2×10^{-9}	1×10^{-9}	2×10^{-10}	
4.8	9×10^{-1}	1×10^{-3}	8×10^{-2}	5×10^{-2}	2×10^{-1}	3×10^{-3}	1×10^{-2}	3×10^{-3}	Overpressure
24	1.0	1×10^{-3}	8×10^{-2}	5×10^{-2}	2×10^{-1}	3×10^{-3}	1×10^{-2}	3×10^{-3}	
					Case AFβ				
0.3	2×10^{-7}	7×10^{-12}	2×10^{-12}	2×10^{-10}	3×10^{-11}	2×10^{-11}	5×10^{-12}	5×10^{-13}	Just Before Melt
2.5	5×10^{-5}	3×10^{-9}	5×10^{-10}	1×10^{-8}	2×10^{-9}	1×10^{-9}	4×10^{-10}	4×10^{-11}	End Melt Rel.
3.5	9×10^{-1}	4×10^{-3}	4×10^{-2}	3×10^{-3}	3×10^{-2}	4×10^{-4}	6×10^{-2}	1×10^{-5}	Steam Expl.
5.5	9×10^{-1}	6×10^{-3}	5×10^{-2}	1×10^{-2}	7×10^{-2}	8×10^{-4}	6×10^{-2}	4×10^{-4}	End Vap. Rel.
24	9×10^{-1}	6×10^{-3}	5×10^{-2}	1×10^{-2}	7×10^{-2}	8×10^{-4}	6×10^{-2}	4×10^{-4}	

TABLE 13 (Continued)

Time hr	Cumulative Fractions of Core Inventory Released to the Atmosphere[a]								Event[b]
	Xe-Kr	Org-I	I-Br	Ca-Rb	Te	Ba-Sr	Ru	La	
					Case AFα				
0.3	$2x10^{-7}$	0	$2x10^{-12}$	$2x10^{-10}$	$3x10^{-11}$	$2x10^{-11}$	$6x10^{-12}$	$6x10^{-13}$	Just Before S.E.
0.3	$3x10^{-1}$	0	$3x10^{-2}$	$2x10^{-2}$	$8x10^{-2}$	$2x10^{-3}$	$1x10^{-1}$	$6x10^{-5}$	Steam Expl.
2.5	$9x10^{-1}$	$2x10^{-3}$	$5x10^{-1}$	$3x10^{-1}$	$2x10^{-1}$	$4x10^{-2}$	$2x10^{-1}$	$1x10^{-3}$	End Melt Rel.
4.5	$9x10^{-1}$	$4x10^{-3}$	$5x10^{-1}$	$3x10^{-1}$	$3x10^{-1}$	$4x10^{-2}$	$2x10^{-1}$	$3x10^{-3}$	
7.2	$9x10^{-1}$	$4x10^{-3}$	$5x10^{-1}$	$4x10^{-1}$	$5x10^{-1}$	$5x10^{-2}$	$2x10^{-1}$	$5x10^{-3}$	After Vap. Rel.
24	$9x10^{-1}$	$4x10^{-3}$	$5x10^{-1}$	$4x10^{-1}$	$5x10^{-1}$	$5x10^{-2}$	$2x10^{-1}$	$5x10^{-3}$	
					Case AEα (wet)				
0.3	$6x10^{-8}$	0	$3x10^{-12}$	$7x10^{-10}$	$1x10^{-12}$	$1x10^{-14}$	0	0	Just Before Melt
2.4	$3x10^{-5}$	$2x10^{-9}$	$2x10^{-8}$	$6x10^{-8}$	$1x10^{-8}$	$8x10^{-8}$	$2x10^{-9}$	$2x10^{-10}$	Just Before S.E.
2.5	$4x10^{-1}$	$3x10^{-3}$	$2x10^{-1}$	$5x10^{-2}$	$3x10^{-1}$	$7x10^{-3}$	$4x10^{-1}$	$2x10^{-4}$	Steam Expl.
3.8	1.0	$7x10^{-3}$	$3x10^{-1}$	$1x10^{-1}$	$4x10^{-1}$	$1x10^{-2}$	$4x10^{-1}$	$2x10^{-3}$	
8.0	1.0	$7x10^{-3}$	$3x10^{-1}$	$2x10^{-1}$	$7x10^{-1}$	$2x10^{-2}$	$5x10^{-1}$	$5x10^{-3}$	After Vap. Rel.
24	1.0	$7x10^{-3}$	$3x10^{-1}$	$2x10^{-1}$	$7x10^{-1}$	$2x10^{-2}$	$5x10^{-1}$	$5x10^{-3}$	
					Case AEγ (dry)				
1.4	$1x10^{-6}$	0	$2x10^{-10}$	$1x10^{-8}$	$9x10^{-11}$	$9x10^{-13}$	0	0	Just Before Melt
2.5	$9x10^{-6}$	$4x10^{-10}$	$8x10^{-10}$	$4x10^{-8}$	$4x10^{-9}$	$3x10^{-9}$	$8x10^{-10}$	$8x10^{-11}$	End Melt Rel.
3.5	$3x10^{-5}$	$2x10^{-9}$	$3x10^{-9}$	$8x10^{-8}$	$2x10^{-8}$	$7x10^{-9}$	$3x10^{-9}$	$3x10^{-10}$	
5.0	$1x10^{-4}$	$7x10^{-9}$	$9x10^{-9}$	$1x10^{-7}$	$1x10^{-7}$	$2x10^{-8}$	$9x10^{-9}$	$1x10^{-9}$	End Vap. Rel.
8.8	$9x10^{-1}$	$6x10^{-3}$	$1x10^{-1}$	$9x10^{-3}$	$1x10^{-2}$	$1x10^{-3}$	$1x10^{-3}$	$2x10^{-4}$	Overpressure
24	1.0	$7x10^{-3}$	$1x10^{-1}$	$9x10^{-3}$	$2x10^{-2}$	$1x10^{-3}$	$1x10^{-3}$	$2x10^{-4}$	
					Case AEβ (dry)				
1.3	$3x10^{-7}$	0	$1x10^{-10}$	$1x10^{-8}$	$9x10^{-11}$	$9x10^{-13}$	0	0	Just Before Melt
2.5	$1x10^{-5}$	$5x10^{-10}$	$8x10^{-10}$	$4x10^{-8}$	$4x10^{-9}$	$2x10^{-9}$	$7x10^{-10}$	$7x10^{-11}$	End Melt Rel.
3.0	$8x10^{-1}$	$6x10^{-3}$	$6x10^{-1}$	$3x10^{-1}$	$9x10^{-2}$	$4x10^{-2}$	$7x10^{-2}$	$1x10^{-3}$	Steam Expl.
5.0	1.0	$7x10^{-3}$	$6x10^{-1}$	$3x10^{-1}$	$1x10^{-1}$	$4x10^{-2}$	$7x10^{-2}$	$2x10^{-3}$	End Vap. Rel.
24	1.0	$7x10^{-3}$	$6x10^{-1}$	$3x10^{-1}$	$1x10^{-1}$	$4x10^{-2}$	$7x10^{-2}$	$2x10^{-3}$	
					Case ADJγ (Release Through Annulus to Atmosphere)				
7.0	$3x10^{-2}$	0	$4x10^{-3}$	$5x10^{-3}$	$4x10^{-5}$	$4x10^{-7}$	0	0	Just Before Melt
8.5	$9x10^{-1}$	$7x10^{-3}$	$7x10^{-2}$	$7x10^{-2}$	$1x10^{-2}$	$1x10^{-2}$	$3x10^{-3}$	$3x10^{-4}$	End Melt Rel.
12.0	1.0	$7x10^{-3}$	$8x10^{-2}$	$9x10^{-2}$	$8x10^{-2}$	$1x10^{-2}$	$7x10^{-3}$	$1x10^{-3}$	End Vap. Rel.
24	1.0	$7x10^{-3}$	$8x10^{-2}$	$9x10^{-2}$	$9x10^{-2}$	$1x10^{-2}$	$7x10^{-3}$	$1x10^{-3}$	

TABLE 13 (Continued)

Time hr	Cumulative Fractions of Core Inventory Released to the Atmosphere[a]								Event[b]
	Xe-Kr	Org-I	I-Br	Ca-Rb	Te	Ba-Sr	Ru	La	
				Case ADJγ (Release Directly to Atmosphere)					
7.0	$3x10^{-2}$	0	$4x10^{-2}$	$2x10^{-2}$	$1x10^{-4}$	$1x10^{-6}$	0	0	Just Before Melt
8.5	$9x10^{-1}$	$7x10^{-3}$	$7x10^{-1}$	$2x10^{-1}$	$5x10^{-2}$	$3x10^{-2}$	$9x10^{-3}$	$9x10^{-4}$	End Melt Rel.
12.0	1.0	$7x10^{-3}$	$8x10^{-1}$	$3x10^{-1}$	$3x10^{-1}$	$3x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	End Vap. Rel.
24	1.0	$7x10^{-3}$	$8x10^{-1}$	$3x10^{-1}$	$3x10^{-1}$	$3x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	
				Case ADFγ (Release Through Annulus to Atmosphere)					
0.3	$5x10^{-3}$	$3x10^{-6}$	$5x10^{-6}$	$6x10^{-5}$	$9x10^{-6}$	$6x10^{-6}$	$2x10^{-6}$	$2x10^{-7}$	Just Before Melt
2.5	$5x10^{-1}$	$3x10^{-4}$	$3x10^{-4}$	$6x10^{-4}$	$1x10^{-4}$	$7x10^{-5}$	$2x10^{-5}$	$2x10^{-6}$	End Melt Rel.
4.0	$8x10^{-1}$	$6x10^{-4}$	$5x10^{-4}$	$2x10^{-3}$	$8x10^{-3}$	$2x10^{-4}$	$5x10^{-4}$	$9x10^{-5}$	
5.5	$9x10^{-1}$	$7x10^{-4}$	$6x10^{-4}$	$3x10^{-3}$	$1x10^{-2}$	$2x10^{-4}$	$7x10^{-4}$	$1x10^{-4}$	End Vap. Rel.
24	1.0	$1x10^{-3}$	$7x10^{-4}$	$3x10^{-3}$	$1x10^{-2}$	$2x10^{-4}$	$8x10^{-4}$	$2x10^{-4}$	
				Case ADFγ (Release Directly to Atmosphere)					
0.3	$5x10^{-3}$	$3x10^{-6}$	$1x10^{-3}$	$1x10^{-3}$	$2x10^{-4}$	$1x10^{-4}$	$3x10^{-5}$	$3x10^{-6}$	Just Before Melt
2.5	$5x10^{-1}$	$3x10^{-4}$	$5x10^{-2}$	$1x10^{-2}$	$2x10^{-3}$	$1x10^{-3}$	$4x10^{-4}$	$4x10^{-5}$	End Melt Rel.
4.0	$8x10^{-1}$	$7x10^{-4}$	$8x10^{-2}$	$2x10^{-2}$	$4x10^{-2}$	$2x10^{-3}$	$3x10^{-3}$	$5x10^{-4}$	
5.5	$9x10^{-1}$	$7x10^{-4}$	$8x10^{-2}$	$2x10^{-2}$	$5x10^{-2}$	$2x10^{-3}$	$3x10^{-3}$	$6x10^{-4}$	End Vap. Rel.
24	1.0	$1x10^{-3}$	$8x10^{-2}$	$2x10^{-2}$	$6x10^{-2}$	$2x10^{-3}$	$4x10^{-3}$	$7x10^{-4}$	
				Case ADEγ (dry) (Release Through Annulus to Atmosphere)					
1.4	$7x10^{-7}$	0	$8x10^{-9}$	$2x10^{-7}$	$1x10^{-9}$	$1x10^{-11}$	0	0	Just Before Melt
2.5	$6x10^{-6}$	$2x10^{-8}$	$5x10^{-8}$	$5x10^{-7}$	$7x10^{-8}$	$5x10^{-8}$	$1x10^{-8}$	$1x10^{-9}$	End Melt Rel.
3.5	$8x10^{-1}$	$5x10^{-3}$	$5x10^{-2}$	$3x10^{-2}$	$4x10^{-2}$	$3x10^{-3}$	$3x10^{-3}$	$5x10^{-4}$	
5.0	1.0	$7x10^{-3}$	$6x10^{-2}$	$4x10^{-2}$	$8x10^{-2}$	$5x10^{-3}$	$5x10^{-3}$	$9x10^{-4}$	End Vap. Rel.
24	1.0	$7x10^{-3}$	$6x10^{-2}$	$4x10^{-2}$	$8x10^{-2}$	$5x10^{-3}$	$5x10^{-3}$	$1x10^{-3}$	
				Case ADEγ (dry) (Release Directly to Atmosphere)					
1.4	$7x10^{-7}$	0	$8x10^{-7}$	$2x10^{-7}$	$1x10^{-9}$	$1x10^{-11}$	0	0	Just Before Melt
2.5	$6x10^{-6}$	$2x10^{-8}$	$5x10^{-6}$	$5x10^{-7}$	$7x10^{-8}$	$5x10^{-8}$	$1x10^{-8}$	$1x10^{-9}$	End Melt Rel.
3.5	$8x10^{-1}$	$5x10^{-3}$	$2x10^{-1}$	$8x10^{-2}$	$1x10^{-1}$	$9x10^{-3}$	$9x10^{-3}$	$2x10^{-3}$	
5.0	1.0	$7x10^{-3}$	$3x10^{-1}$	$1x10^{-1}$	$2x10^{-1}$	$1x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	End Vap. Rel.
24	1.0	$7x10^{-3}$	$3x10^{-1}$	$1x10^{-1}$	$2x10^{-1}$	$1x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	
				Case AGJδ Elevated (Stack) Releases					
4.5	$1x10^{-2}$	0	$6x10^{-8}$	$7x10^{-7}$	$1x10^{-9}$	$1x10^{-11}$	0	0	Just Before Melt
5.5	$6x10^{-2}$	$3x10^{-6}$	$4x10^{-6}$	$4x10^{-5}$	$7x10^{-6}$	$5x10^{-6}$	$1x10^{-6}$	$1x10^{-7}$	End Melt Rel.
6.2	$1x10^{-1}$	$7x10^{-6}$	$8x10^{-6}$	$6x10^{-5}$	$1x10^{-5}$	$8x10^{-6}$	$2x10^{-6}$	$2x10^{-7}$	

TABLE 13 (Continued)

Time hr	Cumulative Fractions of Core Inventory Released to the Atmosphere[a] Xe-Kr	Org-I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La	Event[b]
				Case AGJδ (Continued)					
				Elevated (Stack) Releases (Continued)					
8.2	$3x10^{-1}$	$2x10^{-5}$	$2x10^{-5}$	$1x10^{-4}$	$1x10^{-4}$	$1x10^{-5}$	$9x10^{-6}$	$2x10^{-6}$	End Vap. Rel.
10.1	$3x10^{-1}$	$2x10^{-5}$	$2x10^{-5}$	$1x10^{-4}$	$2x10^{-4}$	$1x10^{-5}$	$1x10^{-5}$	$2x10^{-6}$	
24.0	$6x10^{-1}$	$4x10^{-5}$	$2x10^{-5}$	$1x10^{-4}$	$2x10^{-4}$	$1x10^{-5}$	$1x10^{-5}$	$2x10^{-6}$	
				Ground Level Releases					
4.5	$2x10^{-3}$	0	$6x10^{-7}$	$2x10^{-6}$	$5x10^{-7}$	$5x10^{-11}$	0	0	
5.6	$3x10^{-2}$	$2x10^{-4}$	$2x10^{-4}$	$2x10^{-3}$	$4x10^{-4}$	$2x10^{-4}$	$7x10^{-5}$	$7x10^{-6}$	End 1st Gnd. Leak
6.2	$3x10^{-2}$	$2x10^{-4}$	$2x10^{-4}$	$2x10^{-3}$	$4x10^{-4}$	$2x10^{-4}$	$7x10^{-5}$	$7x10^{-6}$	
8.2	$9x10^{-2}$	$6x10^{-4}$	$6x10^{-4}$	$4x10^{-3}$	$7x10^{-3}$	$4x10^{-4}$	$5x10^{-4}$	$8x10^{-5}$	
10.1	$1x10^{-1}$	$7x10^{-4}$	$8x10^{-4}$	$4x10^{-3}$	$9x10^{-3}$	$4x10^{-4}$	$6x10^{-4}$	$1x10^{-4}$	End 2nd Gnd. Leak
24.0	$1x10^{-1}$	$7x10^{-4}$	$8x10^{-4}$	$4x10^{-3}$	$9x10^{-3}$	$4x10^{-4}$	$6x10^{-4}$	$1x10^{-4}$	
				Case AGJδθ					
				Elevation (Stack) Releases					
4.5	$1x10^{-2}$	0	$6x10^{-6}$	$7x10^{-5}$	$1x10^{-7}$	$1x10^{-9}$	0	0	Just Before Melt
5.5	$6x10^{-2}$	$3x10^{-4}$	$4x10^{-4}$	$4x10^{-3}$	$7x10^{-4}$	$5x10^{-4}$	$1x10^{-4}$	$1x10^{-5}$	End Melt Rel.
6.2	$1x10^{-1}$	$7x10^{-4}$	$8x10^{-4}$	$6x10^{-3}$	$1x10^{-3}$	$8x10^{-4}$	$2x10^{-4}$	$2x10^{-5}$	
8.2	$3x10^{-1}$	$2x10^{-3}$	$2x10^{-3}$	$1x10^{-2}$	$1x10^{-2}$	$1x10^{-3}$	$9x10^{-4}$	$2x10^{-4}$	End Vap. Rel.
10.1	$3x10^{-1}$	$2x10^{-3}$	$2x10^{-3}$	$1x10^{-2}$	$2x10^{-2}$	$1x10^{-3}$	$1x10^{-3}$	$2x10^{-4}$	
24.0	$6x10^{-1}$	$4x10^{-3}$	$2x10^{-3}$	$1x10^{-2}$	$2x10^{-2}$	$1x10^{-3}$	$1x10^{-3}$	$2x10^{-4}$	
				Ground Level Releases					
4.5	$2x10^{-3}$	0	$6x10^{-7}$	$2x10^{-6}$	$5x10^{-9}$	$5x10^{-11}$	0	0	
5.6	$3x10^{-2}$	$2x10^{-4}$	$2x10^{-4}$	$2x10^{-3}$	$4x10^{-4}$	$2x10^{-4}$	$7x10^{-5}$	$7x10^{-6}$	End 1st Gnd. Leak
6.2	$3x10^{-2}$	$2x10^{-4}$	$2x10^{-4}$	$2x10^{-3}$	$4x10^{-4}$	$2x10^{-4}$	$7x10^{-5}$	$7x10^{-6}$	
8.2	$9x10^{-2}$	$6x10^{-4}$	$6x10^{-4}$	$4x10^{-3}$	$7x10^{-3}$	$4x10^{-4}$	$5x10^{-4}$	$8x10^{-5}$	
10.1	$1x10^{-1}$	$7x10^{-4}$	$8x10^{-4}$	$4x10^{-3}$	$9x10^{-3}$	$4x10^{-4}$	$6x10^{-4}$	$1x10^{-4}$	End 2nd Gnd. Rd.
24.0	$1x10^{-1}$	$7x10^{-4}$	$8x10^{-4}$	$4x10^{-3}$	$9x10^{-3}$	$4x10^{-1}$	$6x10^{-4}$	$1x10^{-4}$	
				Case AEGδ (dry)					
				Elevated (Stack) Releases					
1.4	$4x10^{-3}$	0	$3x10^{-7}$	$6x10^{-6}$	$1x10^{-8}$	$1x10^{-10}$	0	0	Just Before Melt
2.5	$2x10^{-2}$	$6x10^{-7}$	$4x10^{-6}$	$4x10^{-5}$	$6x10^{-6}$	$4x10^{-6}$	$1x10^{-6}$	$1x10^{-7}$	End Melt Rel.
3.0	$3x10^{-2}$	$1x10^{-6}$	$7x10^{-6}$	$6x10^{-5}$	$1x10^{-5}$	$7x10^{-6}$	$2x10^{-6}$	$2x10^{-7}$	
5.0	$2x10^{-1}$	$1x10^{-5}$	$3x10^{-5}$	$2x10^{-4}$	$1x10^{-4}$	$2x10^{-5}$	$1x10^{-5}$	$2x10^{-6}$	End Vap. Rel.
7.0	$3x10^{-1}$	$2x10^{-5}$	$4x10^{-5}$	$2x10^{-4}$	$2x10^{-4}$	$2x10^{-5}$	$1x10^{-5}$	$2x10^{-6}$	
24.0	$5x10^{-1}$	$4x10^{-5}$	$5x10^{-5}$	$2x10^{-4}$	$2x10^{-4}$	$2x10^{-5}$	$2x10^{-5}$	$2x10^{-6}$	

TABLE 13 (CONTINUED)

TABLE 13 (Continued)

Time hr	Cumulative Fractions of Core Inventory Released to the Atmosphere[a] Xe-Kr	Org-I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La	Event[b]
				Case AEGδ (dry) (Continued)					
				Ground Level Releases					
3.0	$3x10^{-3}$	$2x10^{-5}$	$8x10^{-5}$	$4x10^{-4}$	$9x10^{-5}$	$6x10^{-5}$	$2x10^{-5}$	$2x10^{-6}$	Start Gnd. Leak
5.0	$6x10^{-2}$	$4x10^{-4}$	$9x10^{-4}$	$4x10^{-3}$	$3x10^{-3}$	$5x10^{-4}$	$3x10^{-4}$	$4x10^{-5}$	
7.0	$8x10^{-2}$	$5x10^{-4}$	$1x10^{-3}$	$5x10^{-3}$	$4x10^{-3}$	$6x10^{-4}$	$3x10^{-4}$	$5x10^{-5}$	End Gnd. Leak
24.0	$8x10^{-2}$	$5x10^{-4}$	$1x10^{-3}$	$5x10^{-3}$	$4x10^{-3}$	$6x10^{-4}$	$3x10^{-4}$	$5x10^{-5}$	
				Case AEGδη (dry)					
1.4	$2x10^{-3}$	0	$2x10^{-5}$	$3x10^{-4}$	$7x10^{-7}$	$7x10^{-9}$	0	0	Just Before Melt
2.5	$7x10^{-3}$	$2x10^{-5}$	$2x10^{-4}$	$2x10^{-3}$	$3x10^{-4}$	$2x10^{-4}$	$5x10^{-5}$	$5x10^{-6}$	End Melt Rel.
3.0	$2x10^{-2}$	$8x10^{-5}$	$4x10^{-4}$	$3x10^{-3}$	$5x10^{-4}$	$3x10^{-4}$	$1x10^{-4}$	$1x10^{-5}$	
5.0	$3x10^{-1}$	$2x10^{-3}$	$4x10^{-3}$	$2x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	$1x10^{-3}$	$2x10^{-4}$	End Vap. Rel.
7.0	$4x10^{-1}$	$3x10^{-3}$	$5x10^{-3}$	$2x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	$2x10^{-3}$	$3x10^{-4}$	
24	$6x10^{-1}$	$4x10^{-3}$	$6x10^{-3}$	$2x10^{-2}$	$2x10^{-2}$	$3x10^{-3}$	$2x10^{-3}$	$3x10^{-6}$	
				Case TCα					
0.8	$5x10^{-1}$	0	$5x10^{-2}$	$4x10^{-3}$	$3x10^{-1}$	$5x10^{-4}$	$4x10^{-1}$	$1x10^{-5}$	Steam Expl.
3.0	$9x10^{-1}$	$3x10^{-3}$	$4x10^{-1}$	$3x10^{-1}$	$3x10^{-1}$	$4x10^{-2}$	$4x10^{-1}$	$1x10^{-3}$	End Melt Rel.
3.9	$9x10^{-1}$	$3x10^{-3}$	$4x10^{-1}$	$3x10^{-1}$	$3x10^{-1}$	$4x10^{-2}$	$4x10^{-1}$	$1x10^{-3}$	Before Vap. Rel.
4.5	1.0	$3x10^{-3}$	$4x10^{-1}$	$4x10^{-1}$	$4x10^{-1}$	$5x10^{-2}$	$5x10^{-1}$	$2x10^{-3}$	1/2 Vap. Rel.
6.0	1.0	$3x10^{-3}$	$4x10^{-1}$	$4x10^{-1}$	$5x10^{-1}$	$5x10^{-2}$	$5x10^{-1}$	$3x10^{-3}$	End Vap. Rel.
12	1.0	$3x10^{-3}$	$4x10^{-1}$	$4x10^{-1}$	$5x10^{-1}$	$5x10^{-2}$	$5x10^{-1}$	$3x10^{-3}$	
				Case TCγ					
3.0	0	0	0	0	0	0	0	0	Vessel Failure
3.2	$9x10^{-1}$	$6x10^{-5}$	$2x10^{-2}$	$4x10^{-2}$	$2x10^{-1}$	$2x10^{-3}$	$1x10^{-2}$	$2x10^{-3}$	Cont. Failure
4.0	$9x10^{-1}$	$6x10^{-5}$	$2x10^{-2}$	$4x10^{-2}$	$2x10^{-1}$	$2x10^{-3}$	$1x10^{-2}$	$2x10^{-3}$	
5.0	1.0	$6x10^{-5}$	$2x10^{-2}$	$6x10^{-2}$	$3x10^{-1}$	$3x10^{-3}$	$2x10^{-2}$	$3x10^{-3}$	End Vap. Rel.
6.2	1.0	$6x10^{-5}$	$2x10^{-2}$	$6x10^{-2}$	$3x10^{-1}$	$3x10^{-3}$	$2x10^{-2}$	$3x10^{-3}$	
15.0	1.0	$6x10^{-5}$	$2x10^{-2}$	$7x10^{-2}$	$3x10^{-1}$	$4x10^{-3}$	$2x10^{-2}$	$4x10^{-3}$	
				Case TWα					
30.0	$4x10^{-1}$	$3x10^{-3}$	$5x10^{-3}$	$4x10^{-2}$	$8x10^{-3}$	$6x10^{-3}$	$2x10^{-3}$	$2x10^{-4}$	1/2 Melt Rel.
31.0	$9x10^{-1}$	$6x10^{-3}$	$1x10^{-2}$	$9x10^{-2}$	$2x10^{-2}$	$1x10^{-2}$	$3x10^{-3}$	$3x10^{-4}$	End Melt Rel.
31.0	$9x10^{-1}$	$6x10^{-3}$	$5x10^{-2}$	$9x10^{-2}$	$2x10^{-1}$	$1x10^{-2}$	$4x10^{-1}$	$3x10^{-4}$	Steam Expl.
32.0	$9x10^{-1}$	$6x10^{-3}$	$6x10^{-2}$	$9x10^{-2}$	$3x10^{-1}$	$1x10^{-2}$	$4x10^{-1}$	$4x10^{-4}$	Vessel Melt
33.5	1.0	$6x10^{-3}$	$7x10^{-2}$	$1x10^{-1}$	$3x10^{-1}$	$1x10^{-2}$	$4x10^{-1}$	$1x10^{-3}$	1/2 Vap. Rel.
35.0	1.0	$6x10^{-3}$	$8x10^{-2}$	$1x10^{-1}$	$4x10^{-1}$	$1x10^{-2}$	$4x10^{-1}$	$2x10^{-3}$	End Vap. Rel.
42.0	1.0	$6x10^{-3}$	$9x10^{-2}$	$1x10^{-1}$	$4x10^{-1}$	$1x10^{-2}$	$4x10^{-1}$	$2x10^{-3}$	

TABLE 13 (Continued)

Time hr	Cumulative Fractions of Core Inventory Released to the Atmosphere[a]								Event[b]
	Xe-Kr	Org-I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La	
				Case TWγ' (Release Directly to Atmosphere)					
30.0	$4x10^{-1}$	$3x10^{-3}$	$4x10^{-1}$	$2x10^{-1}$	$5x10^{-2}$	$3x10^{-2}$	$9x10^{-3}$	$9x10^{-4}$	1/2 Melt Rel.
31.0	$9x10^{-1}$	$6x10^{-3}$	$9x10^{-1}$	$5x10^{-1}$	$9x10^{-2}$	$6x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	End Melt Rel.
32.5	$9x10^{-1}$	$6x10^{-3}$	$9x10^{-1}$	$5x10^{-1}$	$9x10^{-2}$	$6x10^{-2}$	$2x10^{-2}$	$2x10^{-3}$	Vessel Melt
33.0	$9x10^{-1}$	$6x10^{-3}$	$9x10^{-1}$	$5x10^{-1}$	$2x10^{-1}$	$6x10^{-2}$	$3x10^{-2}$	$3x10^{-3}$	1/2 Vap. Rel.
34.5	$9x10^{-1}$	$6x10^{-3}$	$9x10^{-1}$	$5x10^{-1}$	$2x10^{-1}$	$6x10^{-2}$	$3x10^{-2}$	$4x10^{-3}$	End Vap. Rel.
42.0	1.0	$6x10^{-3}$	$9x10^{-1}$	$5x10^{-1}$	$3x10^{-1}$	$6x10^{-2}$	$3x10^{-2}$	$4x10^{-3}$	
				Case TWγ (Release Through Annulus to Atmosphere)					
30.0	$4x10^{-1}$	$3x10^{-3}$	$5x10^{-3}$	$4x10^{-2}$	$8x10^{-3}$	$6x10^{-3}$	$2x10^{-3}$	$2x10^{-4}$	1/2 Melt Rel.
31.0	$9x10^{-1}$	$6x10^{-3}$	$1x10^{-2}$	$9x10^{-2}$	$2x10^{-2}$	$1x10^{-2}$	$3x10^{-3}$	$3x10^{-4}$	End Melt Rel.
32.5	$9x10^{-1}$	$6x10^{-3}$	$1x10^{-2}$	$9x10^{-2}$	$2x10^{-2}$	$1x10^{-2}$	$3x10^{-3}$	$3x10^{-4}$	Vessel Melt
33.0	$9x10^{-1}$	$6x10^{-3}$	$1x10^{-2}$	$9x10^{-2}$	$4x10^{-2}$	$1x10^{-2}$	$5x10^{-3}$	$6x10^{-4}$	1/2 Vap. Rel.
34.5	$9x10^{-1}$	$6x10^{-3}$	$1x10^{-2}$	$1x10^{-1}$	$6x10^{-2}$	$1x10^{-2}$	$6x10^{-3}$	$8x10^{-4}$	End Vap. Rel.
42.0	1.0	$6x10^{-3}$	$1x10^{-2}$	$1x10^{-1}$	$7x10^{-2}$	$1x10^{-2}$	$7x10^{-3}$	$1x10^{-3}$	

(a) Te includes Se and Sb
Ru includes Mo, Pd, Rh, and Tc
La includes Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, and Nb.

(b) Notations used are defined as follows:
Gap Rel. Only - Gap release only
Just Before Melt - Just before core melting begins
End Melt Rel. - Melt release is complete
End Vap. Rel. - Vaporization release is complete
After Vap. Rel. - After vaporization release has ended
Overpressure - Containment failure by overpressure
Steam Expl. - Containment failure by steam explosion
Just Before S.E. - Just before a steam explosion occurs
End Gnd. Leak - Ground level leakage complete.

TABLE 14 SUMMARY OF HAND CALCULATED RESULTS FOR BWR SEQUENCES

Case	Time, hr	Cumulative Core Fraction Lost to Atmosphere[a]			
		I,Br	Org-I	Te	Ru
AGJεζ	4.2	∿0	∿0	∿0	0
	5.7	$2x10^{-2}$	$4x10^{-3}$	$1x10^{-2}$	$2x10^{-3}$
	8.5	$4x10^{-2}$	$7x10^{-3}$	$2x10^{-1}$	$1x10^{-2}$
AGJεη	4.2	∿0	∿0	∿0	0
	5.7	$7x10^{-3}$	$3x10^{-3}$	$7x10^{-3}$	$1x10^{-3}$
	8.5	$8x10^{-3}$	$7x10^{-3}$	$4x10^{-2}$	$3x10^{-3}$
AGJε	4.2	∿0	∿0	∿0	0
	5.7	$1x10^{-5}$	$2x10^{-5}$	$1x10^{-5}$	$2x10^{-6}$
	8.5	$5x10^{-5}$	$6x10^{-5}$	$2x10^{-4}$	$2x10^{-5}$
	24.0	$7x10^{-5}$	$1x10^{-4}$	$5x10^{-4}$	$4x10^{-5}$
AFGεζ	2.5	$4x10^{-3}$	$2x10^{-3}$	$8x10^{-4}$	$2x10^{-4}$
	4.0	$9x10^{-3}$	$4x10^{-3}$	$2x10^{-2}$	$2x10^{-4}$
	8.5	$2x10^{-2}$	$4x10^{-3}$	$2x10^{-1}$	$9x10^{-3}$
AFGεη	2.5	$2x10^{-3}$	$2x10^{-3}$	$7x10^{-4}$	$1x10^{-4}$
	4.0	$6x10^{-3}$	$2x10^{-3}$	$1x10^{-3}$	$2x10^{-4}$
	8.5	$2x10^{-2}$	$4x10^{-3}$	$2x10^{-1}$	$9x10^{-3}$
AFGε	2.5	$4x10^{-6}$	$4x10^{-6}$	$1x10^{-6}$	$2x10^{-7}$
	4.0	$7x10^{-6}$	$1x10^{-5}$	$2x10^{-6}$	$5x10^{-7}$
	8.5	$2x10^{-5}$	$1x10^{-4}$	$6x10^{-4}$	$3x10^{-5}$
	24.0	$4x10^{-5}$	$2x10^{-4}$	$2x10^{-3}$	$1x10^{-4}$

(a) Te includes Se and Sb
Ru includes Mo, Pd, Rh, and Tc.

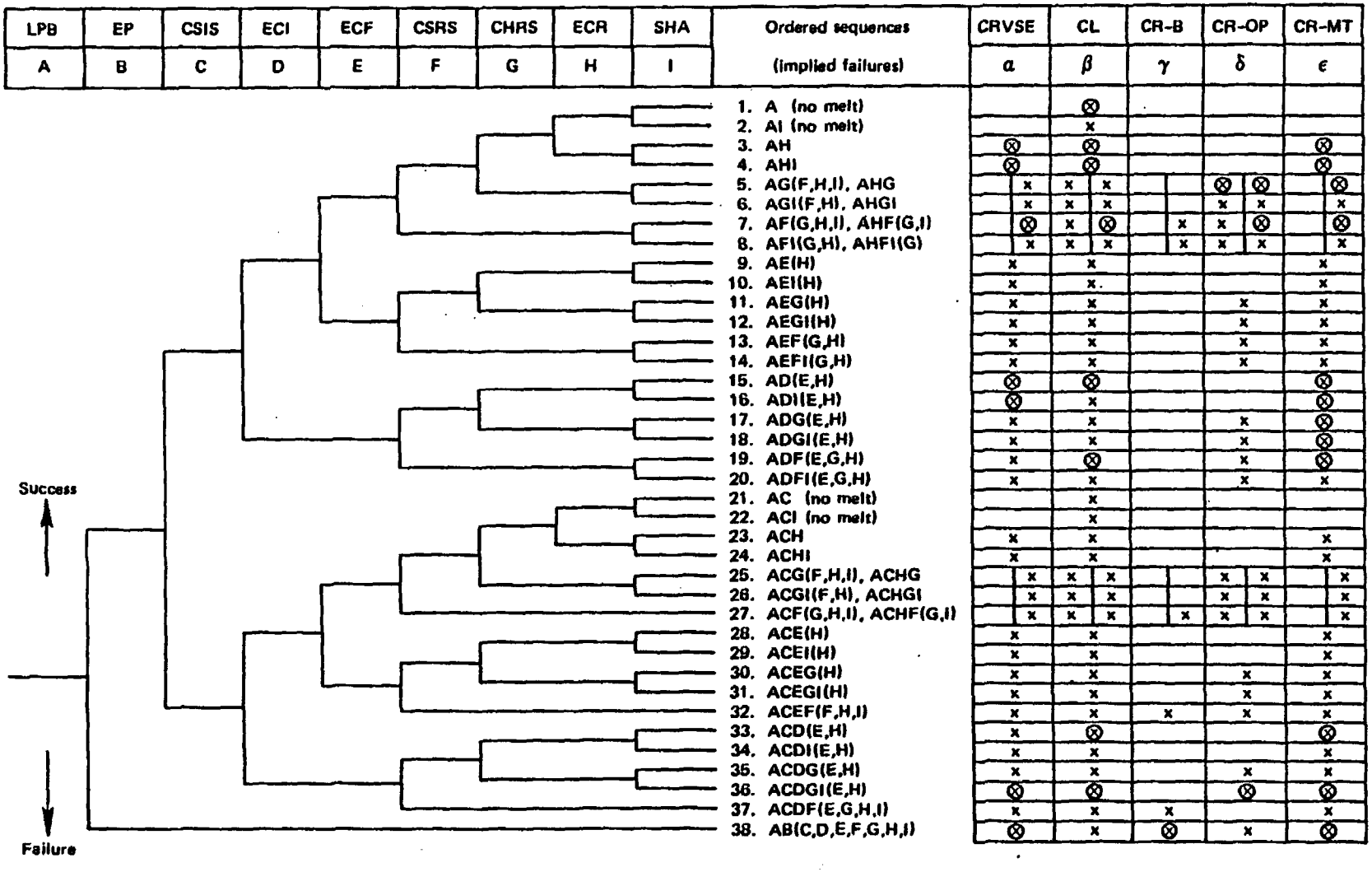


FIGURE 1. PWR LARGE LOCA EVENT TREE

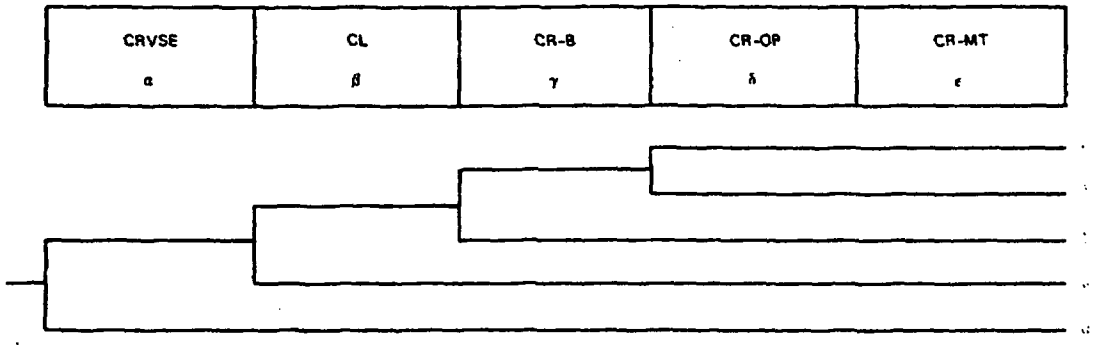
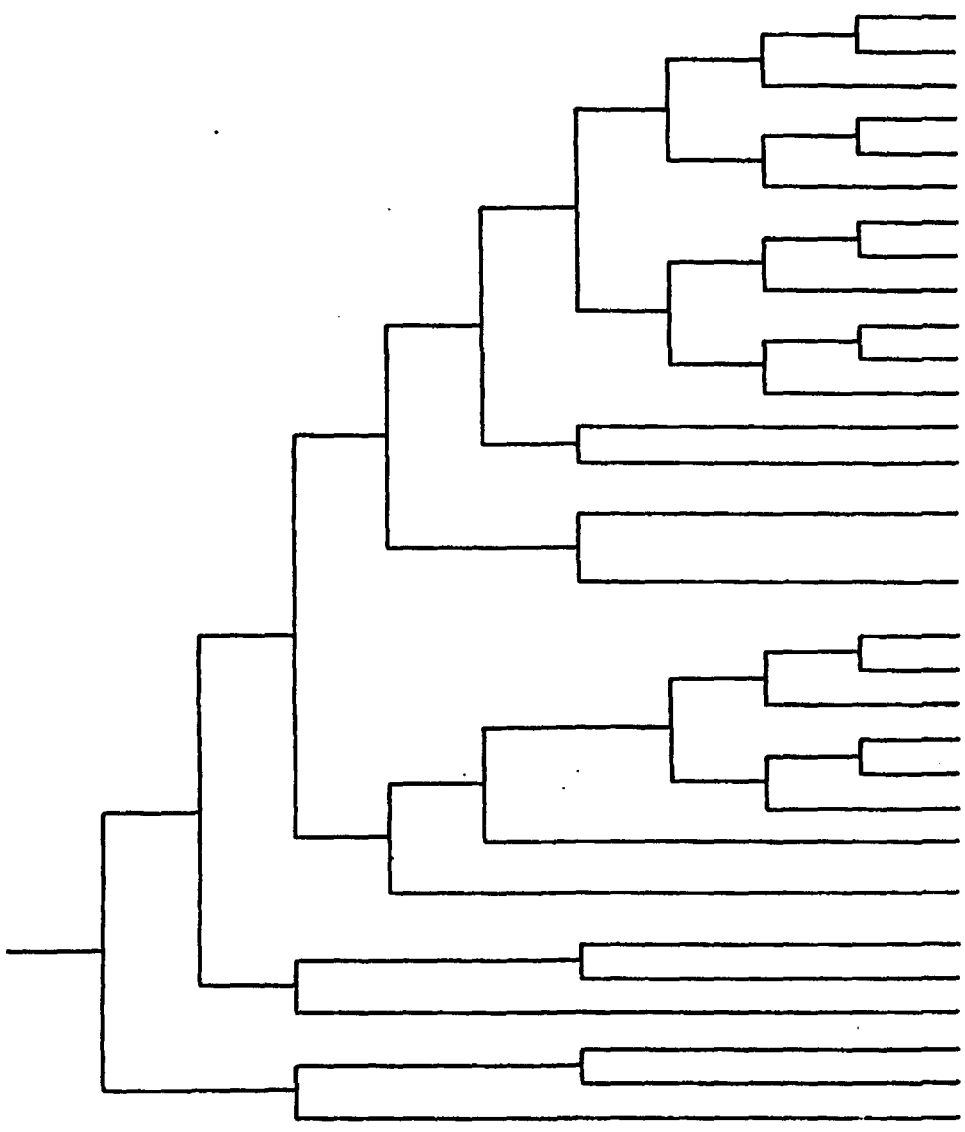


FIGURE 2. PWR CONTAINMENT EVENT TREE



PB	EP	Scram	VS	ECO	ECF	CL	CSRS	LPCRS	HPSW	Sequence	Containment failure modes										
A	B	C	D	E	F	G	H	I	J		α	β	γ	$\epsilon\zeta$	$\epsilon\eta$	$\epsilon\theta$	ϵ	$\delta\zeta$	$\delta\eta$	$\delta\theta$	δ
										1. A											
										2. AJ	×		⊗′								
										3. AI	×		×								
										4. AH											
										5. AHJ	×		×								
										6. AHI	×	×	×								
										7. AG											
										8. AGJ	×	×		⊗	⊗	×	⊗	×	×	⊗	⊗
										9. AGI	×	×		×	×	×	×	×	×	×	×
										10. AGH											
										11. AGHJ	×	×		×	×	×	×	×	×	×	×
										12. AGHI	×	×		×	×	×	×	×	×	×	×
										13. AF	⊗	⊗	⊗								
										14. AFG	×	×		⊗	⊗	×	⊗	×	×	×	×
										15a. AE	⊗	×	×								
										15b. AE		⊗	⊗								
										16a. AEG	×	×		×	×	×	×	×	×	×	×
										16b. AEG		×		×	×	×	×	×	⊗	×	⊗
										17. AD											
										18. ADJ	×		⊗′								
										19. ADI	×		×								
										20. ADH											
										21. ADHJ	×		×								
										22. ADHI	×		×								
										23. ADF	×		⊗′								
										24a. ADE	×		×								
										24b. ADE			⊗′								
										25. AC	×	×	×								
										26. ACG	×	×		×	×	×	×	×	×	×	×
										27. ACD			×								
										28. AB		×	×								
										29. ABG		×		×	×	×	×	×	×	×	×
										30. ABD			×								

FIGURE 3. BWR LARGE LOCA EVENT TREE

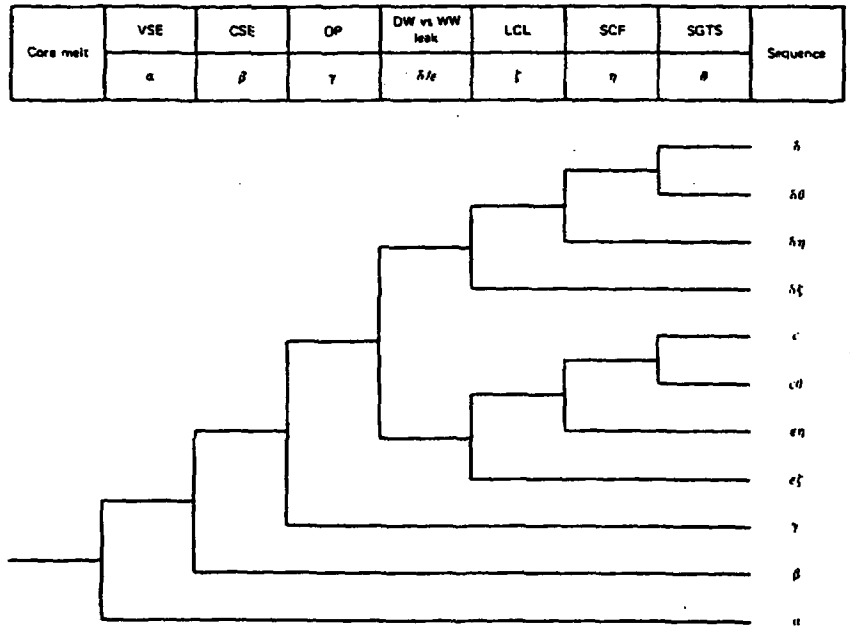


FIGURE 4. BWR CONTAINMENT EVENT TREE