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A Large Dry PWR Containment Response Analysis for Postulated Severe Accidents

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가상적 중대사고에 대한 대형건식 가압경수로 격납용기의 반응해석

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Abstract

A large dry PWR containment response analysis for postulated severe accidents was performed as part of the Zion Risk Rebaselining study for input to the U.S. NRC's "Reactor Risk Reference Document," NUREG-1150. The Methodologies used in the present work were developed as part of the Severe Accident Risk Reduction Program(SARRP) at Sandia National Laboratory specifically for the Surry Plant, but they were extrapolated to Zion. Major steps of the quantification of risk from a nuclear power plant are first outlined. Then, the methodologies of containment response analysis for severe accidents used for Zion are described in detail: major features of the containment event tree(CET) analysis codes and CET quantification procedures are summarized. In addition, plant specific features important to containment response analysis are presented along with the containment loading and performance issues included in the present uncertainty analysis. Finally, a brief summary of the results of deterministic and statistical containment event tree analysis is presented to provide a perspective on the large dry PWR containment response for postulated severe accidents.

요 약

미국 원자력 안전규제위원회(U.S. NRC)의 "원자로 위험도 참고문서(NUREG-1150)"에 입력자료로 제공하기 위해 실시한 Zion발전소 안전성 재평가 작업의 일환으로, 가상적 중대사고에 대한 대형건식 가압경수로 격납용기 반응해석을 수행하였다. 본 연구에서 사용한 방법론들은 Sandia 국립연구소에서 "중대사고 위험도 감소계획"의 일환으로 특히 Surry 발전소에 대한 연구를 위해 개발한 것이며, 이 방법론을 Zion발전소에 외삽법으로 적용하였다. 먹저, 원자력발전소의 위험도를 정량적으로 평가하는 주요절차를 개설하였다. 그리고, Zion발전소의 중대사고에 대한 격납용기 반응해석을 위해 사용한 방법론들을 상세히 기술하였다. 즉, 격납용기 반응해석을 위해 사용한 방법론들을 상세히 기술하였다. 즉, 격납용기 사건수목 해석 전산코드의 주요 특징과 격납용기 사건수목의 정량적 평가절차를 요약하여 높액다. 격납용기 반응해석에 있어서 중요한 발전소 고유의 특성과 본 연구의 불확실성 분석에 포함시킨 격납용기 하중과 성능에 관계되는 문제점들을 아울러 제시하였

다. 끝으로, 가상적 중대사고에 대한 대형건식 가압경수로 격납용기의 반응에 대한 전망을 제공하기 위해서 결정적 및 통계학적 격납용기 사건수목 해설결과를 간단히 요약하여 제시하였다.

1. Introduction

In the period directly following the TMI-2 accident in 1979, a major research was undertaken by the U.S. NRC to review existing U.S. Plant designs and identify potential risks to the public from those reactor accidents that would be beyond the coverage of present licensing design basis events. In addition, major programs were initiated by the U.S. NRC, in particular, to develop an improved understanding of severe accidents and to establish a technical basis to support regulatory decisions (1, 2).

More recently, the response of containments to severe accidents at several commercial nuclear plants, such as Zion (3), Surry (4, 5) and Sequoyah (6) has been analyzed under the Severe Accident Risk Reduction Program (SARRP). These results as well as the previous PRA's show that the greatest impact on the amount of radioactivity released from a plant is determined by whether the containment fails or not, or if it fails, the timing of the failure. Therefore, the containment response analyses for postulated severe accidents focus on identifying the various pathways that could lead to the release of fission products beyond the containment boundaries and on estimating their frequency of occurrence.

A large dry PWR containment response analysis for postulated severe accidents to be presented here is mainly based on the results and insights gained from the containment response analysis performed at the Brookhaven National Laboratory (BNL) as part of the Zion Risk Rebaselining study (3) for input to the U.S. NRC's "Reactor Risk Reference Document" NUREG-1150 (1). The updating of risk for Zion was based on the methodology developed as part of the SARRP as Sandia National Laboratoy (SNL) (4-6). This methodology was developed at SNL specifically for Surry plant (a Westinghouse-derigned three-loop PWR reactor with a subatmospheric large dry containment) but was extrapolated to Zion (a Westinghouse-designed four-loop PWR reactor with a large dry containment).

The main objectives of this paper are to present:

(1) the "Containment Event Tree Approach" used in the Zion analysis (3),

(2) the major uncertainty issues associated with the containment loading and performance, and

(3) results of the point-estimate and the statistical containment response analysis for a large dry PWR containment (3).

In addition, this work is directed to assisting analysts in evaluation of PWR containment response for postulated severe accidents as an independent work and/or as part of an overall task such as evaluation of severe accident risks and the potential for risk reduction for any nuclear power station.

2. Quantification of Risk from a Nuclear Power Plant

Assessment of risk from the operation of nuclear power plants involves determination of the likelihood of various accident sequences and their potential offsite consequences. The risk from a nuclear power plant can be defined by

$$R^K=\sum_i f_i \sum_j C_{ij}\ r^K_j(S_{ij}) \tag{1}$$

where

R^K=risk of type K(associated with consequence K),

f_i=frequency of plant damage state(PDS)$_i$,

C_{ij}=conditional probability of containment release category j given PDS i (i.e., containment matrix),

S_{ij}=fission product source term for containment release category j of PDS i,

$r_K{}^j$=consequence of type K, given fission product source term (S_{ij}), for release category j.

Thus, there are five distinct but closely related phases in the risk analysis of nuclear power plant operation. It should be noted here that the matrix formulation is a powerful tool in a systematic quantification of the overall risk and also in identification of the dominant contributors to the overall risk. The matrix approach lends itself easily to computerization, thus improving calculational accuracy, productivity, and documentation. It is also very useful in carrying out sensitivity analysis (7). The framework of the matrix approach is already established in Reference (8), and also in detail in Reference (9). A more detailed description of the steps of the matrix analysis is given in Reference (7). The major steps of the matrix analysis for quantification of risk are summarized in Table 1 (7). Present study focuses on the methodology used to obtain C_{ij} in Eq. (1) (or C_{jk} in Table 1) and the result of the containment response analysis for severe accidents for Zion (3).

Table 1. Major Steps of the Matrix Analysis for Quantification of Risk (7)

Step	Description	Result
1	Construct the vector of initiating event frequencies:	$\hat{a}$
2	Construct the "plant systems matrix" of conditional probabilities: m_{ij}=conditional probability of PDS j given initiator i	M
3	Calculate the vector of core melt state (i.e., PDS) frequencies: $\hat{f}=M\hat{a}$	$\hat{f}$
4	Construct the containment matrix of conditional probabilities: C_{jk}=conditional probability of release category K given PDS j.	C
5	Calculate the vector of "release category frequencies": $\hat{r}=C\hat{f}=CM\hat{a}$	$\hat{r}$
6	Construct a "site matrix" for each of the damage categories: S_{Km}=consequence m given release category K	S_i
7	Draw a risk curve for each damage category.	risk curve

3. Methodology of Containment Response Analysis for Severe Accidents Used for Zion

In traditional probabilistic risk assessments (PRAs), the accident pathways that contribute to risk are described by two types of event trees: (1) "system event trees" are used to define the spectrum of accident sequences (i.e., the combinations of accident initiators and subsequent system failures) that can lead to core melt, (2) "containment event trees (CET)" are used to define the containment failure modes which lead to fission product release beyond the containment boundary.

However, the current method of a PWR containment response analysis for postulated severe accidents include steps 1-4 shown in Table 1. According to the current CET quantification method used in the Zion (3) and Surry (4,5), the step 4 in Table 1 can be further divided into 5 detailed steps as shown in Table 2.

The major objective of the containment response analysis is to determine, given a core-melt accident, when and how the containment conditions could affect a release. Therefore, the containment response analyses for postulated severe accidents focus on identifying the various pathways that could lead to the release of fission products beyond the containment boundaries and on estimating their frequency of occurrence.

Once the PDS frequencies, f_i in Eq. (1), are obtained from the step 3 in Table 1, the

Table 2. Major Step of the Current CET Quantification Method Given the PDS Frequencies f_i

Step	Description	Result
1	Construction of the plant-specific phenomenological containment event tree in the form of computer codes.	CET codes (e.g., EVNTRE or EVNTREISS)
2	Thermal-hydraulic analysis of containment loading by codes (e.g., MARCH and CONTAIN), estimation of branch point probabilities to prepare the plantspecific input data for CET, and assessment of data for dependency upon prior events.	Branch point probability data, dependency data.
3	Characterization of containment release modes (i.e., containment bins which result in source term bins).	Binning data for CET code
4	Identification of containment loading and performance issues for uncertainty analysis, and determination of sample data and weighting factors for statistical sampling.	Issue data, sample data, and weighting factors
5	Quantification of the CET matrix C_{ij} via CET codes such as EVNTRE (10) and EVNTREISS (10) for deterministic and statistical containment response analysis, respectively.	Containment matrix C_{ij}
(Note)	For risk reduction analysis, in particular, "Identification of the Potential Accident-Prevention and Accident-Mitigation Options" should be added as an additional step and the step 5 should be repeated.	Modified containment matrix C_{ij}

next step for quantification of risk is to extend the analysis to determine the containment response to these damage states. For this purposes a phenomenological containment event tree(step 1 in Table 2) is constructed. The core melt states (i.e., PDSs) become the initiator for this event tree. Mass and energy balances in the containment after a core melt event are analyzed by codes such as MARCH (11) and CONTAIN (12) (step 2 in Table 2) and failure frequencies are assigned to various phenomena that may take place in the containment. The event sequences resulting from core melt states are quantified and are sorted into radioactive fission product release categories to be used in the site-specific public damage analysis (step 3 and 5 in Table 2). For uncertainty analysis, in particular, the step 4 in Table 2 should be performed before step 5. A containment matrix is then constructed (step 5 in Table 2). Each element in this matrix is the conditional frequency of occurrence of a radioactive release category given the occurrence of a core melt initiating event. A more detailed description of the major steps of the current CET quantification method is given here and in the next sections. All the examples to be shown here are adopted from the result of the Zion analysis (3).

(1) Accident Sequence Evaluation and Characterization of the PDSs

For the purpose of present work calculational details to obtain PDS frequencies f_i are omitted: a brief outline of the method used to characterize PDSs is reviewed here in the context of the interface requirements of the containment and source term phases of the analysis.

For the purpose of interfacing with the accident progression analysis, the accident sequences are grouped into plant damage states. The grouping is effected in such a way that all sequences within a group are essentially equivalent with regard to accident progression. A plant damage state is labelled by up to four letters in the Zion study (3, 13). The first letter represents the initiating event (or the primary system states up to core damage):

A≡large or medium LOCA, core damage at low pressure,

S≡small LOCA, core damage at high pressure,

T≡transient initiator, reactor coolant system (RCS) remains intact until core damage,

V≡interfacing systems LOCA.

The second letter represents the timing of

Table 3. Plant Damage States (3, 13)

Accident Sequence Class	Representative Accident Sequence for Reactor Calculatons	Containment Safeguards Status		PDSs (Containment Event Tree Designators)
		Fan Coolers	Sprays	
I	Large LOCA	1	1	AEFC
	ECCS failure	1	0	AEF
	at injection	0	1	AEC
		0	0	AE
II	Large LOCA	1	1	ALFC
	ECCS failure	1	0	ALF
	on Recirc.	0	1	ALC
		0	0	AL
III	Small LOCA	1	1	SEFC
	ECCS failure	1	0	SEF
	on injection	0	1	SEC
		0	0	SE
IV	Small LOCA	1	1	SLFC
	ECCS failure	1	0	SLF
	on Recirc.	0	1	SLC
		0	0	SL
V	Transient with	1	1	TEFC
	no Cooling	1	0	TEF
		0	1	TEC
		0	0	TE
	V Sequence	—	—	V

core melt:

E≡early core melt (ECC failure in the injection phase),

L≡late core melt (ECC failure in the recirculation phase).

The presence of either of the last two letters (i.e., the third and fourth letters) indicate that the fan cooler or containment spray systems operate successfully:

F≡success of containment fan coolers,

C=success of containment spray injection or recirculation.

Absence of either of these last two letters indicate complete loss of containment heat removal.

In order to provide a complete framework for containment response analysis, it is necessary to consider for each of the five accident sequence classes (shown in Table 3) four possible combinations of fan cooler system and containment spray system operation:

1. Fans on, sprays off;
2. Sprays on, fans off;
3. Both sprays and fans on;
4. Both sprays and fans off.

These combinations lead to the defination of 20 distinct plant damage states. In addition, the interfacing systems LOCA (the V sequence) constitutes the twenty-first state. The V sequence is assumed a priori to bypass the containment (therefore, V sequence does not challenge containment) and thus it is not treated by the containment event tree (3). In Table 3, the PDSs and representative accident sequences are defined (13). Table 4 displays the PDS frequencies over the 100 samples (3), whereas Reference (14) provides explicit calculations of the reference case.

Table 4. Damage State Mean Frequencies for 100 Latin Hypercuve Vector Samples (3)

Damage State	Mean Frequency
AEC	7.0×10^{-7}
TEC	1.3×10^{-6}
SEFC	9.9×10^{-6}
TEFC	1.3×10^{-6}
AEFC	7.0×10^{-7}
SEC	7.9×10^{-6}
SE	5.6×10^{-7}
ALFC	7.3×10^{-6}
SLFC	8.7×10^{-6}
ALF	0
SLF	0
ALC	7.2×10^{-6}
SLC	8.6×10^{-6}
AL	3.4×10^{-7}
SL	2.9×10^{-7}
V	1.0×10^{-7}
Mean Total Comulative Damage Frequency	5.5×10^{-5}

(2) Major Features of the CET Analysis Codes Used and CET Quantification Procedures

A major focus in SARRP was the development of a containment event tree (CET) for each of the plant types represented by the reference plants (4, 5). These event trees delineated the various physical processes and phenomena of a core melt accident and the pathways that could lead to release of fission products as a consequence of core-damage accidents. Its purpose is to examine the response of the containment and to establish the most likely modes of containment failure.

In the Zion study (3) the CET in the form of computer codes (such as EVNTRE and EVNTREISS) provided the necessary framework for quantification of the likelihood of various containment failure modes. These CET codes have been developed by the SNL as part of the SARRP program (3). Except for the detailed input data, the Zion CET code is the same as that of Surry (4, 5). The plant-specific containment event tree was developed by identifying the types of containment responses that might be expected to impact risk and the various events and conditions that could affect those responses, at a level of detail that could reasonably be supported by the information currently available (4). This led to the construction of an event tree that is significantly expanded be yond those previously used in PRAs: the structure of the Zion CET is based on 59 top events, many of which have more than two branching options. The 59 top events shown in Table 3-3 of Reference (3) can be classfied into 6 distinct group of questions as shown in Table 5. These top event questions are posed in ways that require the answers to be expressed in terms of likelihoods.

The major features of the Zion CET codes (i.e., EVNTRE and EVNTREISS) can be summarized ac follows:

1. Zion CET codes allow for nodes in the CET with arbitrary number of branches, which differs from a typical event tree that has only two branches "yes" or "no".
2. The branch point probabilities can be dependent upon branches taken at prior nodes.
3. Information can be fed to the program in the form of "parameters" which the program can manipulate at a later node to internally calculate a branch point probability based upon user input criterion.
4. The outcomes of the event tree are binned by input specified by the user.

The Zion CET codes allow the user to develop very complex event trees; ones which could never actually be drawn on paper. The containment matrix C_{ij} in Eq. (1) can be quantified by either of the code, EVNTRE (10) or EVNTREISS (10). The basic difference between EVNTRE and EVNTREISS is that the former is a deterministic code, while the latter is a statistical code with some input from LHS 77

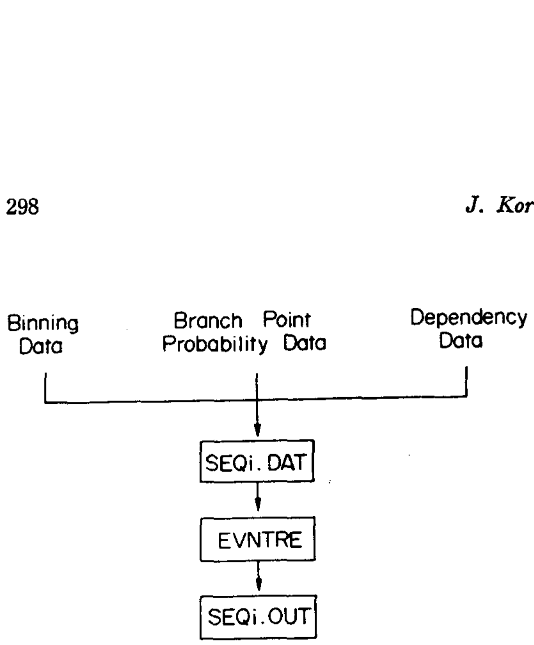

Fig. 1. Schematic Diagram for EVNTRE Calculation (3).

(15). Actually EVNTREISS is a modified version of the EVNTRE code and the EVNTREISS code allows one to step through all issues in a Latin Hypercube sample. For actual deterministic quantification of the containment matrix EVNTRE code is used along with three different types of input data, namely (1) binning data, (2) branch-point probability data and (3) dependency data as shown in Fig. 1 (3).

The branch point probabilities shown in Fig. 1 are a set of probabilities for each branch of the top event. These probabilities are conditional and depend on the previous branch(or branches). Dependency information and branch point probabilities should be provided for an EVNTRE (or EVNTREISS) analysis. The information may be provided by an expert (or a group of experts) who is familiar with the containment event tree or the accident phenomenology. Some of the branch point probabilities need to be calculated internally. In this case, "parameters," instead of the probabilities themselves, are input to calculate branch probabilities. This is most frequently the case for the probability of containment failure. In these instances, pressure loads are compared to estimates of a containment failure pressure distribution (estimated using "parameters") to obtain an estimate of containment failure probability. Depending on the types of input, there are six different types of top events and they are shwon in Table 6.

After marching through the event tree and finding the probability for each of the pathways, all of the different outcomes may be condensed into a more tangible form. The binning data shown in Fig. 1 is used to bin all of the different outcomes in a user specified fashion. For example, each pathway of the event tree may be binned according to its source term characteristics, i.e., release category. Finally, using "SEOi. DAT" (i.e., the CET input data prepared for PDS *i*), an EVNTRE calculation is performed to obtain SEOi. OUT which consists of binning probabilities. This procedure is repeated for every plant damage state to obtain the containment matrix.

For statistical quantification of the containment matrix, on the other hand, EVNTREISS code is used with (1) Issue data and (2) Sample data in addition to the three input data required for EVNTRE. For those issues in the containment event tree judged to most likely have a significant impact on risk ranges of plausible values were developed, and, for each parameter value fractional weights were assigned by experts to provide an estimate of the validity associated with that particular value. This process of combining the use of complex containment event trees and associated expert judgement quantification where data were lacking was instituted for NUREG-1150 (1) since it is particularly suitable for the study of incompletely understood physical processes such as severe accident progressions. This process was also used to combine accident sequence frequency, containment performance, and source term uncertaintiies into overall risk uncertainties in the NUREG-1150 (1).

In the Zion analysis (3), the different sets of values for branches of each issue are called "levels". The levels may represent different opinions about the severities of a certain physical phenomenon. However, the probability of occurrence of each "level" can be different from each other. The "weighting factor" for example, can be considered as this subjective probability of occurrence of each level. The weighting factors are used in LHS77 (**15**) as the SUBROUTINE USRDIST. The combination of one issue with other issue is done according to the weighting factors. For example, there are two issues with 4 and 3 levels for each issue, respectively, as shown in Fig. 2. The weighting factors are (0.1, 0.4, 0.3, 0.2) for Issue 1 and (0.33, 0.34, 0.33) for Issue 2. Assume that we have 30 samples using LHS77. Each sample consists of the level numbers of Issue 1 and Issue 2. Fig. 2 shows the combination process by LHS77. The last column under LHS samples shows how Issue I levels are combined with Issue 2 levels.

(3) Characterization of Containment Release Modes (Containment Bins which Result in Source Term Bins)

The outcomes of the containment event tree form a set of discriminated accident pathways. The combination of a given accident sequence and a given path through the CET is called a "scenario". Such a scenario provides a description of the initial and boundary conditions required to assess the resulting source term. Since the numer of such pathways (i.e., scenarios) for each core melt sequence is large, it is necessary to combine these scenarios into a smaller set of groups, called bins, which are judged to be similar in terms of parameters considered to be important to the source terms. Examples of parameters included in defining bins are:

1. reactor coolant system thermal-hydraulic conditions during meltdown,
2. timing and mode of containment failure,
3. availability of containment engineering safeguard features.
4. state of the debris after vessel breach with respect to coreconcrete interactions.

That is, the release categories are defined so that the end states from the CET paths can be assigned on the basis of risk-related characteristics (radionnclide inventory as well as the time, duration, and energy of a potential release). For Zion (**3**). the containment release modes are classified into 19: For reference purposes, the bins are delineated by characteristics in Table 7. The definnition of the source term bins used in the Zion study (**3**) are similar to

Table 5. Classification of the 59 CET Top Events (3, 4)

Classification of Top Events	Questions Addressed
(1) Entry States:	The conditions (1) in the reactor coolant system and (2) containment prior to melting of the core that could influence the accident progression.
(2) Phenomenological Events:	(1) The physical phenomena that could affect the progression of severe accidents, (2) time frame of occurrence, and (3) their subsequent effects on the accident development.
(3) Reactor Coolant System Failure Modes:	(1) The size and location of the reactor coolant system breach and (2) the corresponding pressure during core meltdown.
(4) Survivability of Containment Safeguards Systems:	(1) Whether the containment fan cooler and spray systems survive the conditions occuring during severe accidents that exceed their design bases.
(5) Active System Status and Recovery:	(1) Availability of AC power after the initiating event, whether AC has been restored after vessel breach and/or before or after 6 hours, and (2) operability of ECCS.
(6) Containment Failure Modes:	(1) The loads that challenge containment, and the survivability of the containment to these loads, (2) the nature of the failure (size and location), and (3) the subsequent pathways for release of fission products to the environment.

Level	Weighting Factor	LHS Sample
Level 1	0.1	1
		1
		1
Level 2	0.4	2
		2
		2
		2
		2
		2
		2
		2
		2
		2
		2
		2
Level 3	0.3	3
		3
		3
		3
		3
		3
		3
		3
		3
Level 4	0.2	4
		4
		4
		4
		4
		4

Issue 1

Level	Weighting Factor	LHS Sample
Level 1	0.33	1
		1
		1
		1
		1
		1
		1
		1
		1
		1
Level 2	0.34	2
		2
		2
		2
		2
		2
		2
		2
		2
		2
Level 3	0.33	3
		3
		3
		3
		3
		3
		3
		3
		3
		3

Issue 2

Level No. of Issue 1	Level No. of Issue 2
1	2
4	1
2	1
2	3
4	3
2	1
2	2
3	2
2	2
2	2
3	3
4	2
3	3
3	2
2	3
1	2
1	1
3	1
3	3
4	1
3	1
2	3
3	3
2	3
2	1
2	1
4	2
3	1
4	2
2	3

LHS SAMPLES

Fig. 2. Level Combinations between Two Issues

that given in NUREG-0956 (16) for Surry, with additional bins assigned to direct heating resulting from high pressure ejection, namely, bins 16, 17, 18 and 19. Bins 1~4 and 16~19, respectively, are equivalent except that bins 16~19 include effects due to direct heating, and bins 1~4 do not. Bins 18 and 19 are introduced for the Limited Latin Hypercube (LLH) sampling study. Bin 13 is used for a pressurized containment (no sprays); bin 14, for unpressurized (sprays on). Bins 7 and 6 are for isolation failure sequences with and without sprays, respectively.

Bins for interfacing-system LOCA(V-sequence,

Table 6. Six Types of CET Top Events

	Dependency Upon Prior Events	
	Independent	Dependent
Branch Point Probabilities only	Type 1	Type 2
Branch Point Probabilities and Parameter Values	Type 3	Type 4
A Set of Parameters to be Summed and Compared to Reference Parameters to Obtain Branch Point Probabilities	Type 5	Type 6

Table 7. Characteristics of the Source Term Bins

Sequence and Containment Status	Source Term Bins																		
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
Containment Failure																			
Rupture Before Core Melt					√														
Early Overpressure	√	√	√	√												√	√	√	√
Late Overpressure-Rupture							√	√											
Late Overpressure-Leakage										√									
Melt-through													√	√					
Leak or Isolation Failure						√	√												
Containment Bypass-SGTR											√								
Containment Bypass-Dry												√							
No Failure															√				
Containment Spray System																			
(See Note 1) Operates		√		√			√	√						√			√		√
Fails	√		√			√			√				√			√		√	
Primary System Pressure																			
High	√	√														√	√		
Moderate			√	√														√	√
Low																			
Containment Pressure																			
(See Note 2) High													√						
Low														√					
Water Available to Cavity																			
Yes		√		√				√											
No	√		√		√	√	√		√							√	√	√	√
Direct Heating Effect																			
None	√	√	√	√															
Significant																√	√	√	√

√—The characteristic is required for the bin. Characteristics not maked are not determinant of the bin and any combination may apply.

Note 1: The spray question is also dependent on timing. Critical time frames are different for different bins.

Note 2: This is only used as a discriminator for Bins 13 and 14. Obviously, containment pressure is high in many other bins and these are not checked.

bin 11 and bin 12) were slightly modified in order to account for induced LOCA situations: bin 12 now includes all V-sequences. Bin 11 is for induced LOCA at steam generator (steam generator tube rupture).

For point-estimate containment response analysis, the likelihood of the 15 release modes (source term bins) was calculated for each of the 8 PDSs. For statistical LLH containment analysis, on the other hand, the likelihood of the 19 source term bins was calculated for each of the 14 PDSs. This generated the frequencies for each of the source term bins, for integration into the risk calculation.

4. Plant Specific Features Important to Containment Response Analysis

The brance points and probabilities for the Zion containment event tree (3) reflect consideration of a number of plant-specific features that could have important effects on the progression of a severe accident. More detailed description of the Zion plant-specific features regarding the containment failure pressure containment safeguard systems, and the reactor cavity geometry is given below to put present analyses in right perspectives.

(1) Containment Pressure Capacity

The pressure at which failure would be expected to occur is a key determinant in the likelihood and timing of containment failure during pлessure transients that result from the generation of steam or noncondensibles, direct heating and the combustion of hydrogen. The pressure at containment failure can also influence the dispersion of fission products released to the enviroment.

The detailed description of the Zion containment is qiven is Reference (17).

(2) Containment Safeguard Systems

The containment fan cooler and spray systems provide redundant and diverse containment heat removal capability for Zion. The initiation pressure for fan coolers and sprays are 0.14 MPa (19.7 Psia) and 0.26 MPa (37.1 Psia), respectively (17). More detailed description can be found in Reference (17).

(3) Reactor Cavity Geometry

The progression of the accident, following the reactor pressure vessel (RPV) failure is strongly affected by the reactor coolant syatem pressure before RPV failure, the cavity geometry, and water availability. The design of the reactor cavity, in particular, can have an important impact on the accident progression due primarily to: (1) the degree to which the core debris is inhibited from being dispersed following ejection from the reactor vessel, (2) the ability of water in containment to reach the core debris, and (3) the ability to transfer heat from the cavity to the containment atmosphere. The major characteristics of the Zion reactor cavity geometry is given in Reference (3).

5. Containment Loading and Performance Issues Included in the Zion Uncertainty Analysis

During a core melt accident, there are several possible types of containment loads that could occur. In each type of the containment loads, there still remains significant uncertainty. The uncertainty analysis approach relies on the selection of key uncertainty issues that can have a significant impact on the estimated risk at Zion. The approach used in the selection and evaluation of key uncertainty issues for Zion containment loading and performance is basically the same as that used in the analysis of containment response for Surry (4). Thus, as in the case of Surry (4), the containment loading and performance issues shown in Table 8 were also included in the application of the Limited Latin Hypercube (LLH) approach (15) to Zion (3).

The issues shown in Table 8 were discussed in detail for Surry in Reference (4), For the present work, only a brief outline of the issues is given here for reference (3).

Issue 1 : The probability and location of an induced failure of the RCS pressure boundary can be an important consideration in the determination of the timing and mode of containment failure. This issue influences the potential for primary system depressurization during core meltdown which in turn influences the potential for a high pressure melt ejection and containment failure due to direct heating by the dispersed core debris.

Issue 2 : The mode of reactor pressure vessel breach is another important consideration, which can affect the early containment failure.

Issue 3 : The magnitude of pressure loading of the containment at vessel breach is strongly governed by the direct heating phenomenology.

Direct containment heating phenomenon involves the possible heating and pressurization of the containment atmosphere due to dispersal of core debris following ejection of melt from the vessel under high pressure. It is potentially capable of generating extremely large containment loads, but the actual loads to be expected from it are very uncertain, with negligible loads also being considered possible (**18**). The Zion specific input data for this issue were derived by extrapolating the values suggested for Surry.

There are five levels which represent a wide spectrum of pressure load due to steam spike and direct containment heating issues. The respective pressure increment and, weighting factor for each level are listed in Table 9. There are two major important factors which determine the magnitude of pressure increment, namely; RCS pressure at the time of vessel breach, and existence of water in the reactor cavity. Depending on these conditions different values of multipliers are assigned. The conditions and multipliers are listed in Table 10.

Issue 4 : During a core melt accident, significant quantities of H_2 and other combustible gases could be generated. If these combustible gases accumulated to large concentrations before igniting, the resulting deflagration could impose a high pressure and temperature loads on the containment. The magnitude of hydrogen burn pressure increment at vessel breach includes all of the uncertainties associated with hydrogen production and combustion issues.

Issue 5 : The containment failure pressure is the most important issue, because the failure pressure is the key parameter in determining the timing and likelihood of the containment failure, during overpressure accident conditions that result from the substantial release of steam and other gases into the containment, or other events such as hydrogen combustion and direct containment heating. The input data for each level of failure pressure were obtained; by extrapolating the Zion plant specific data along with a multiplier based on the rate of Zion to Surry containment pressure capacities. For each level, the containment failure pressure and the corresponding standard deviation are shown in Table 11.

Issue 6 : The size of containment failure is also considered important because of two principal correlations associated with this issue, namely (1) the size of failure is correlated with the ultimate pressure capacity, and (2) the size of failure is also dependent on the level of pressurization relative to the mean failure pressure.

Issue 7 : The probability of containment spray failure resulting from the consequences of the accident condition is also considered to be important because there are a number of uncertainties associated with the spray failure due to core debris accumulation in the contai-

Table 8. Containment Loading and Performance Issues for the Uncertainty Study

Issue No.	Subject of the Seven Issues
1	Probability and Location of Induced Failure of the RCS Pressure Boundary
2	Mode of Reactor Vessel Breach
3	Magnitude of Pressure Loading at Vessel Breach due to Direct Heating and Steam Spike
4	Magnitude of Hydrogen Burn Pressure Increment at Vessel Breach
5	Containment Failure Pressure
6	Size of Containment Failure
7	Probability of Late Containment Spray System Failure

Table 9. Pressure Increase Due to Direct Heating and Steam Spike

Level	Pressure Increment	Weighting Factor
1	20 psi	0.18
2	55 psi	0.32
3	85 psi	0.23
4	115 psi	0.16
5	140 psi	0.11

Table 10. Conditions and Multiplier

Conditions		Multiplier
RSC Pressure	Cavity	
High	Dry	1.0
High	Full	0.76
Intermediate	Dry	0.76
Intermediate	Full	0.50
Low	Dry	0.0
Low	Full	0.0

Table 11. Containment Failure Pressure and Standard Deviation

Level	Failure Pressure	Standard Deviation	Weighting Factor
1	110 psia	17.4	.20
2	149 psia	21.7	.48
3	175 psia	26	.25
4	215 psia	17.4	.07

nment sump or damage caused by hydrogen combustion. This issue is directly related to the operability of the active containment engineered safety features.

6. Containment Response Analysis Results and Discussion

A Brief summary of the results of deterministic and statistical containment event tree analysis is presented here to provide a perspective on the large dry PWR containment response for postulated severe accidents.

(1) Deterministic CET Analysis Results

For the point-estimate, the branch probability of direct containment heating was assumed to be zero. It should be noted here that, for the Zion plant, the initial operation of the containment sprays will lead to a flooded cavity with a correspondingly higher potential for ex-vessel debris coolability.

The results of the point-estimate calculations are given in Table 12 in terms of conditional frequencies of containment failure for the various accidents that were assessed to be major contributors to core melt at Zion (a summary is given in Tables 2.2 and 3.2 in References 3). It must be noted that the containment matrix shown in Table 12 ignores the contributions due to Bins 16 through 19 (direct containment heating failure modes).

Table 12 shows that the point-estimate calculations indicate that for most accident sequences, there is a high likelihood that containment integrity will be maintained. However, containment failure will occur with a relatively high likelihood for the plant damage state 'SE' through a variety of failure mechanism; with late overpressurization being the dominant failure mode.

(2) Statistical CET Analysis Results

The results of the containment uncertainty analysis using the LLH approach are shown in Fig. 3 and 4 and Table 13. Fig. 3 and 4 show the median, 5th, and 95th percentile values of the conditional probability for early (Bins 1~5 and 16~19) and late (Bins 8~10) containment failure probabilities, respectively. Early containment failure is taken here to include all cases for which significant radionuclide releases to the environment occur before or during the time of reactor vessel breach. Table 13 shows some statistical parameters for all containment failure modes. These LLH results were obtained by the application of the EVNTREISS (10) computer code that has incorporated the LLH sampling technique for the key uncertainty issues of the containment loading and performance shown in Table 8.

It should be noted here that the number of LLH samples was limited to 100. The observations that can be made from the results presented here are, therefore, limited by the sample size. However, valuable insights concerning the LLH approach can be gained by simply comparing the LLH results with that of the point-estimate.

Table 12. The Containment Matrix (Point-Estimate)

	SEFC	SLFC	TEFC	AEFC	ALFC	SEC	SE	V	SGTR
Bin 1							1.8-4		
2			1.0-4						
3									
4	1.0-4*	1.0-4		1.0-4	1.0-4	1.0-4			
5									
6							2.5-3		
7	2.5-3	2.5-3	2.5-3	2.5-3	2.5-3	2.5-3			
8	3.3-3	3.3-3	3.3-3	2.0-3	2.9-3	3.3-3			
9							4.54-1		
10							3.08-1		
11									1.0
12								1.0**	
13							23.5-1		
14									
15	9.94-1	9.94-1	9.94-1	9.94-1	9.94-1	9.94-1			

* $1.0\text{-}4 = 1.0\times10^{-4}$

** The Break location is assumed to be above water, i.e., no scrubbing.

Thus, the following observations can be made from results displaysed in Fig. 3 and 4 as well as in Table 13.

1. Fig. 3 shows that the LLH results for early containment failure probability range from 0.0 and 0.76 for the plant damage state TEFC, while it varies from 1.0×10^{-4} to nearly unity for SE. For the remaining plant damage states (SEFC, AEFC, and SEC) the early containment failure probability ranges are slightly narrower than that for TEFC. The LLH median values are from one to two orders of magnitude larger (i.e., higher probability for early containment failure) than the point-estimates. The pointestimates have very low probability of early containment failure (1.0×10^{-4} to 1.8×10^{-4}) for all PDSs and they are close to the bottom of the LLH distribution.

2. Fig. 4 indicates that the LLH results for late containment failure probability ranges from

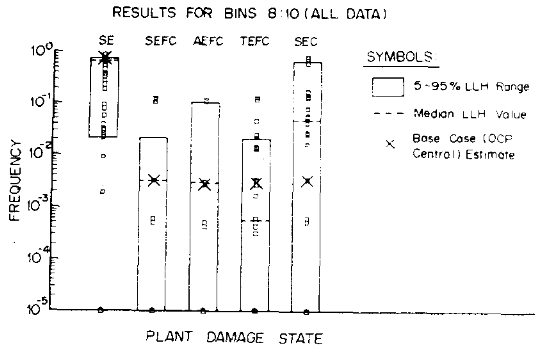

Fig. 3. Comparison of LLH and Point-Estimate Results for Conditional Probability of Early Containment Failure (Bins 1~5 and 16~19) **for Five Representative Plant Damage States.**

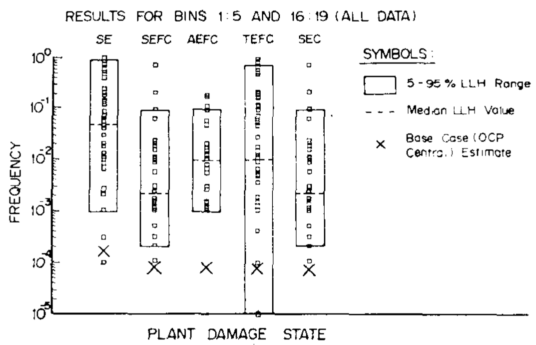

Fig. 4. Comparison of LLH and Point-Estimate Results for Conditional Probability of Late Containment Failure (Bins 8~10) **for five representative Plant Damage States.**

Table 13. Some Statistical Parameters for the 100 LLH Samples of the Conditional Probability of Containment Failure Modes (All Sequences Included)

	Minimum	5th	Median	95th	Maximum	Mean
Early Failure	3.4-4	1.0-3	1.1-2	0.17	0.66	4.0-2
Late Failure	3.5-5	3.5-3	1.9-2	0.16	0.54	5.9-2
Isolation Failure	2.0-4	1.1-3	2.5-3	2.5-3	2.5-3	2.3-3
Melt through	0.0	4.3-5	1.9-3	8.1-3	4.4-2	3.0-3
V	5.3-4	6.6-4	2.7-3	1.8-2	0.14	6.8-3
No Failure	0.30	0.52	0.94	0.98	0.99	0.89

0.0 to about 0.77 for SEC, while the ranges for other PDSs are slightly narrower. This figure also shows that the point-estimates for the conditional probability of late containment failure are similar to the median values of the LLH results for all PDSs.

3. Table 13 shows some statistical parameters derived from all LLH samples of the conditional probability of each containment failure mode. According to Table 13, the most dominant failure mode is late overpressurization failure with the mean value of about 6 percent, followed by early failure with mean conditional probability of 4 percent. These two overpressurization failures have a conditional probability of 0.1, which is about 50% lower than the early containment failure probability reported for the Surry plant.

(3) Discussion of the Results

Major factors that have contributed to the difference between the point-estimates and the LLH results are as follows (3):

1. The direct containment heating phenomenon was absent in the pointestimate analysis whereas the effect of the direct containment heating phenomenon was addressed as a major uncertainty issue for the containment loading in the LLH samples. Therefore, the LLH results for early containment failure probability shown in Fig. 3, in particular, include effects due to direct heating and this factor brought about a shift in the early containment failure probability.

2. The pressure loads associated with hydrogen burn were smaller in the point-estimate analysis than for the average LLH input.

3. The point-estimate analysis has one containment failure pressure distribution while the LLH sampled with lower failure pressures.

Thus, a comparison of the point-estimate and LLH results tends to indicate that the LLH uncertainty band, particularly for early containment failure, appears to be mostly in the upward direction, and the LLH result seems to be dominated by containment failures resulting from direct containment heating. The main reason for this result is attributable to the fact that the LLH input from the expert reviewers favored a more severe containment condition due to direct heating scenarios.

7. Summary and Conclusion

The containment response analysis results of the Zion nuclear power plant (3) are summarized here to provide a perspective on the large dry PWR containment response for postulated severe accidents:

1. The probability for early containment failure when considering all accident sequences is plotted in Fig. 5 along with the mean, the median and the uncertainty range represented by the 5th and 95th percentiles. As shown, the uncertainty ranges from 1.0×10^{-3} to 0.173. The median value is 1.1×10^{-2} and the mean

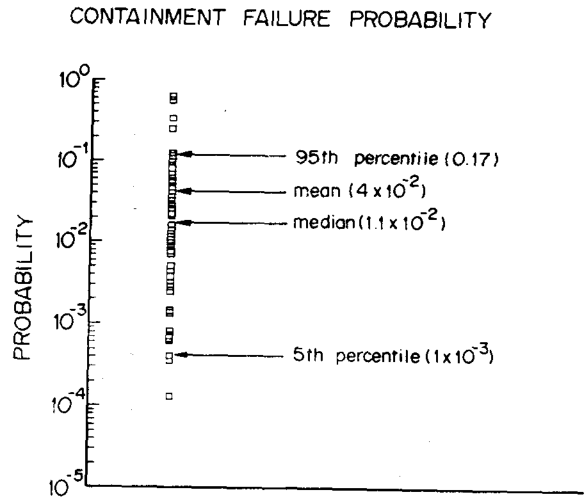

Fig. 5. Conditional Probability of Early Containment Failure: All Sequences included.

value is 4.0×10^{-2}.

2. The Zion results are compared with other studies in Table 14. The Surry study has a mean value of 0.2 which is about an order of magnitude higher than that for Zion. The IDCOR analysis predicted a significantly lower probability of 5×10^{-3} based on the assumption that early containment failure is dominated by the isolation failure. A separate calculation for the isolation failure probability for the Zion containment showed a similar result to that of the IDCOR analysis. The mean probabilities of isolation failure were 2.3×10^{-3} in the present study and 5.0×10^{-3} in the IDCOR analysis, respectively.

3. Fig. 6 is reproduced directly from the NUREG-1150 (1). In this figure, the ranges of the estimated conditional probabilities of early containment failure, weighted by core damage frequency, are displayed for 5 plants including the Zion containment analyzed in the present work. The horizontal lines within the vertical bars represent the individual sample results from the uncertainty analysis and provide a qualitative indication of the concentration trends within the range, based on the judgement of

Table 14. Comparison of Early Containment Failure Probability with Other Studies.

	5th	95th	Median	Mean
Zion	1.0×10^{-3}	0.173	1.1×10^{-2}	4.0×10^{-2}
Surry*	1.5×10^{-2}	0.50	0.1	0.2
RSS*	—	—	—	0.2
IDCOR*	—	—	—	5×10^{-3}

* Data from the Surry Report.

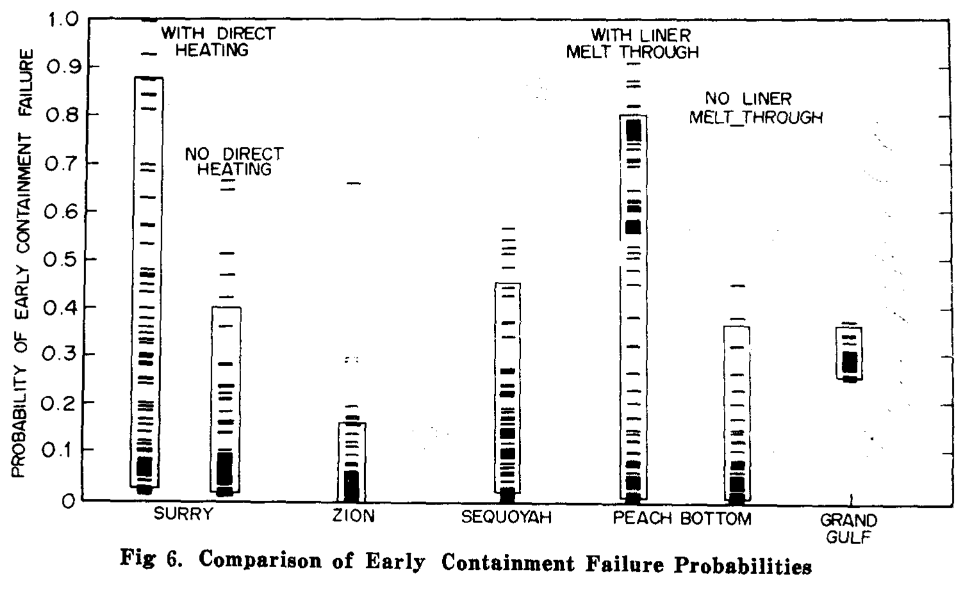

Fig 6. Comparison of Early Containment Failure Probabilities

experts. For the large, dry PWR containment such as Zion (3) and Surry (4, 5), the majority of results tend to indicate a relatively low probability of early failure: however, the containment failure probability could be high if the pressure increments from direct heating and hydrogen burning are near the high ends of their ranges. Thus, for those plants with a large, dry PWR containment it appears that the probability of failure is relatively low, but the potential for a high likelihood of failure cannot be completely ruled out.

Finally, it should be noted that the risk has not been calculated here, only the frequencies of accident pathways that lead to fission product source terms have been estimated. Evaluation of risk requires two additional steps: (1) estimation of source terms for all important sequences and accident pathways, and (2) determination of the mean consequences associated with each source term.

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References

1. Executive Summary of the "Reactor Risk Reference Document," NUREG-1150, U.S. Nuclear Regulatory Commission, 1986.
2. D.F. Ross and G.P. Marino, "NRC Severe Accident Research Program," Presented at the EPRI Information Meeting on the EPZ and the Source Term held at the Brookhaven National Laborotory, Jun., 1986.
3. M. Khatib-Rahbar, C. Park, M. Chun et al., "Evaluation of Severe Accident Risks and Potential for Risk Reduction: Zion Power Plant," NUREG/CR-4551, BNL/NUREG-52029, Vol. 5, for U.S. Nuclear Regulatory Commission, Dept. of Nuclear Energy, Brookhaven National Laboratory, Feb., 1987.
4. A.S. Benjamin, D.M. Kunsman, D.C. Williams et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Surry Power Station, Unit 1," Sandia National Laboratory, SAND 86-1309, Draft, May, 1986.
5. A.S. Benjamin, V.L. Behr, F.E. Haskin et al., "Containment Event Analysis for Postulated Severe Accidents at the Surry Nuclear Power Plant," Sandia National Laboratory, SAND 86-0119, Preliminary Draft for Review, Jan., 1986.
6. A.S. Benjamin, D.M. Kunsman, D.C. Williams et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Sequoyah Power Station, Unit 1," NUREG/CR-4551, SAND 86-1309, Vol. 2, Sandia National Laboratory, Jul., 1986.
7. Selim Sancaktar, "An Illustration of Matrix Formulation for a Probabilistic Risk-Assessment Study," Lecture Notes, Integrated Safety Analysis of Nuclear Power Plants: Methodology and Applications, A 5 Day Course, Sponsored by the Dept. of Nuclear Engineering, KAIST, pp. 5. 80-5. 90, Jul., 1987.
8. Stanley Kaplan, "Matrix Theory Formalism for Event Tree Analysis: Application to Nuclear-Risk Analysis," Risk Analysis, Vol. 2, pp. 9-18, 1982.
9. "Zion Probabilistic Risk Assessment Study," Commonwealth Edison Company, 1981.
10. "Computer Codes EVNTRE, EVNTREISS, and Input Data for Zion S_3D and Surry Nuclear Power Plants," Provided by Sandia National Laboratories, Albuquerque, NM 87185, Feb., 1986.
11. MARCH Code: Derived from Appendix VIII of

"Reactor Safety Study", WASH-1400, USNRC, 1975.

12. K.D. Bergeron et al., "User's Manual for CONTAIN 1.0: A Computer Code for Severe Nuclear Reactor Accident Containment Analysis," SAND 81-1495, NUREG/CR-2224, Sept., 1984.
13. "Zion Probabilistic Safety Study," Commonwealth Edison Company, Sept., 1981.
14. T.A. Wheeler, "Anlysis of Core Damage Frequency from Internal Events: Zion Unit 1," NUREG/CR-4550, Vol. 6, Sandia National Laboratories, Albuquerque, NM 87185, Jul., 1986.
15. R.L. Iman et al., "A FORTRAN 77 Program and User's Guide for the Generation of Latin Hypercube and Random Samples for Use with Computer Models," NUREG/CR-3624, Mar., 1984.
16. M. Silberberg et al., "Reassessment of the Technical Bases for Estimating Source Terms," NUREG-0956, U.S. Nuclear Regulatory Commission, Jul., 1985.
17. "Zion Station Final Safety Analysis Repor," Commonwealth Edison Company, Sept., 1981.
18. D.C. Williams and J.M. Griesmeyer, "Risk-Significance of Chemical Issues Affecting the Source Term," Presented at the American Chemical Society Severe Accident Chemistry Symposium held at Anaheim, California, SAND 86-2021C, Sandia National Laboratory, Sept., 1986.