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INTERNATIONAL TECHNOLOGY CORPORATION

March 9, 1987

Dr. Themis P. Speis
Director, Division of Safety Review and Oversight
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Speis:

<u>Submittal of Revisions to the IDCOR Individual Plant Evaluation (IPE) Pressurized Water Reactor Systems Methodology</u>

We have made minor revisions to the various portions of the IDCOR IPE methodology. The revision pages for the IDCOR PWR systems methodology for review by your staff are enclosed. Please provide the enclosed revisions to your staff as needed.

We consider the PWR systems methodology to be final with the integration of these revisions. A user guide will be prepared for this methodology that will be similar to the BWR methodology users guide provided to you in a previous submittal. We expect to have this available in April.

If you have any questions concerning these revisions, please call Jim Carter at (615) 481-3300.

Sincerely,

James C. Carter for

Anthony R. Buhl
IDCOR Program Manager

ARB:JCC:gf
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1.2 APPLICABILITY METHODOLOGY - GENERIC PROGRAM

The general approach is similar to that of the Byron Risk Study. A Probabilistic Risk Evaluation based on the Zion Probabilistic Safety Study. The main exceptions to the approach will be a deemphasis of Steam Generator Tube Rupture (SGTR) issues and an added emphasis on dealing with system variations. This decision is based on the following data. The results of the Byron Risk Study and subsequent industry risk evaluations have concluded that SGTR initiators are minor core damage risk contributors.

Although USNRC generic issues with respect to SGTR phenomena post core exist, these have little impact on the programs purpose in identifying specific outliers. Thus few added insights would be gained from such a treatment. The level of methodology detail for SGTR generic applicability is consistent with the methodology proposed herein. Next the Byron plant is very similar to the Zion design. Thus, the applicability of Zion analysis to Byron systems was readily apparent for most cases, although several systems required unique analysis. However, there is much more variability of system designs when all PWRs are considered. Although many variations exist, the same basic safety functions are utilized as well as similar train orientations of support functions. These variations are rather limited due to the functional constraints. Thus, the program will have more emphasis on identifying and analyzing these variations.

In general, this program is divided into two main segments. The first is to develop a methodology which rebaselines the reference plants into standardized segments of analysis. From this rebaselined effort, "known" dominant sequences are identified for specific analysis. These include the loss of recirculation after LOCAs and the V sequence. The intent of this segment is to compare the specific plant frequency with frequencies of the base plants. The second segment involves the search for design or operational outliers. This segment does not include the generation of a Level I PRA but rather the use of representative unavailability quantification modules referred to as "TEMPLATES". These templates coupled with generic event trees as well as data and support system checklists will form the baseline for the outlier screening. The intent is to develop reliability building blocks which can be customized to the plant specific designs and can be recombined into the sequences as may be important.

Appendices B.1-B.4 provide example notebooks as well as blank standardized forms (worksheets and tables) for assisting the analyst and for documenting the assessment. In order for consistency across all IPE studies to be achieved, these notebooks must be completed with at least as much information in them as in the examples.

In IPE studies it is imperative that quality assurance techniques be applied due to the large amounts of input/output data generated and also because the analysis is used in support of licensing and safety decision making. Because of the inherent nature of a IPE, it is not possible and as stated by the USNRC not required to apply the exact same type of Quality Assurance (QA) prescribed by 10CFR50 Appendix B because of the following three considerations:

1. The purpose of IPE is to analyze the "best estimate" response of safety and non-safety related systems and structures to a wide range of accidents beyond the design basis. This differs substantially from the process of classical safety analysis in which very high confidence levels are sought to meet regulatory criteria.

2. Certain phenomenology modeled in containment analysis codes is based on best engineering judgement and not on experimental data. Hence, these codes could not be benchmarked against test data.

3. Certain component failure rates are based on best engineering judgment in lieu of actual statistics. This is required because certain failure modes have never occurred and hence there are no statistics.

Despite these limitations, the use of subjective judgement of knowledgeable experts is appropriate and necessary to understand the expected response. The QA program used by utilities or their contractors for IPE studies should include the following items:

A. If computer codes are used during the program, they must be configuration controlled and documented in recorded calculations (departmental calcnotes). These calcnotes are subject to an independent review and sign-off process.

B. All hand calculations are documented (via calcnotes), independently reviewed and checked for accuracy.

C. The results of computer calculations are independently reviewed to verify proper modeling and use of the computer code.

D. Areas in which subjective judgement is used are independently reviewed for reasonableness and consistency.

E. Areas in which data were manually transferred from the output of one code to the inputs to another are independently reviewed and checked for accuracy.

F. Permanent legible records of all analyses are maintained. These are written as departmental calcnotes that require compliance to a review and sign-off procedure prior to acceptance.

In summary, the IPE Quality Assurance program chosen must insure that all computer codes (if used) are independently reviewed and documented, all results are independently reviewed and documented, and all input/output data or engineering assumptions are documented and traceable.

2.4 QUANTIFICATION DATA

The purpose of this section is to demonstrate applicability of referenced point estimates of the failure rates for system components to plant-specific phenomena. These point estimates are necessary for the quantification of the risk models (event trees and fault trees). A methodology is developed in order to determine applicable point estimate failure values for components. This methodology consists of three phases: 1) generic data bank selection; 2) plant specific data collection and, 3) comparison of generic and plant-specific estimates and determination of final values for use in the quantification. The procedural steps for the analysis are described in the following subsections. Appendix B.4 provides examples.

2.4.1 GENERIC FAILURE RATE DATA BANK SELECTION

This section provides two generic data bases for use in the comparison to plant specific values. Table 2.4-1A lists the component types, failure modes, medians, and means from the Interim Reliability Evaluation Program (IREP) Procedures Guide (NUREG/CR-2728). A lognormal distribution is assumed for failures of the components with a 90% probability interval. An upper bound at the 95th percentile, a lower bound at the 5th percentile and error factor are also listed in the table. This data base applies to all systems. The second data base (Table 2.4-1B) is to be applied only in the case of system and fault tree template fits to Zion and Oconee PRAs. This table lists the generic data base information used in the Zion and Oconee PRAs plus the plant-specific failures, trials and Bayesian-updated means for basic components of several systems. A lognormal distribution (90% confidence interval) is assumed. The upper bound at the 95th percentile, median, mean, and error factor are reported in the table. All failure rates are in units of failure per hour while the demand failure probabilities represent failures per demand of the component.

Although this section provides two generic data bases for use in the comparison to plant specific values, it is recognized that other NRC

approved data banks exist, (i.e., IEEE-500, NUREG/CR 2815, ASEP, etc.). Other NRC approved data banks may be used to quantify the unavailability of failed components for system analyzed using the methodology put forth in this document. However, when so used it should be noted and referenced in the summary report and system notebooks of the plant evaluated. It should further be noted that if other than the IREP data base is used, the data applicability curves presented in this section are not applicable to other data bases used to test plant specific data.

accident sequences which identify the initiating event and the progression of the event to core damage through identification of the frontline and support system failures;.2) list of the core damage frequency by plant damage state, and definition of each damage state, and the frequency of the damage state, 3) the table from the impaired containment Screening analysis (Section 2.2.5.1) and any vulnerabilities that were idenlified, and 4) the identification of potential interfacing systems LOCA locations and their frequency of occurrence. This information provides the link between the plant analysis and the containment analysis.

TASK 10: COMPARISON OF PLANT RESULTS TO IDCOR RESULTS

The results from Task 7 for individual plant evaluated will be compared to the generic results of the IDCOR reference plant analyses to determine if major differences exist (Section 4.0). Where these differences do exist a more detailed investigation of the causes will occur.

TASK 11: PREPARATION OF FINAL SUMMARY REPORT

At the completion of the IPE analysis (tasks 1 through 10) the utility must provide a summary report documenting the overall analysis and results. Therefore, the purpose of this task is to summarize the findings and summarize the results of each of the individual tasks into a single integrated document for reporting the IPE results. A suggested "table of contents" for the final summary report is provided in table 3-1. The summary report provides an overview of the IPE objectives, motivations, scope, results, and conclusions. The final report envisioned should be in the range of 75 to 100 pages including all text material, tables and figures. Additional appendices providing backup analysis, documentation, etc. can be added at the discretion of the utility.

The key documentation section of the IPE summary report is Section 3, Plant Analysis, and should contain the following information:

A. A brief discussion of the dominant core damage accident sequences. Dominant sequences can be defined as those sequences that contribute greater then 2.0E-6 or 5% of the total core damage frequency. As a secondary measure, the sum of the reported dominant accident sequences should account for at least 70 to 80 percent of the total core damage frequency. Table 3-2 provides an example as to the level of detail to be included in the discussion.

B. A brief discussion summarizing the core damage frequency by initiating event, plant damage state, and support state including tables. Example tables are shown in by tables 3-3 and 3-4.

C. A brief discussion of the plant's system unavailabilities calculated in task 7. The discussion the should include a brief comparison of the plant's system unavailabilities to the system unavailabilities calculated for similar systems in pervious PRAs.

TASK 12: INCORPORATION OF COMMENTS

Comments received on IDCOR IPE results during the finalization of the summary report will be resolved and incorporated as necessary.

TABLE 3-1 <u>IPE SUMMARY REPORT TABLE OF CONTENTS</u>

TABLE 3-2 DOMINANT ACCIDENT SEQUENCE DISCUSSION

Sequence 1: Small LOCA - Failure of Recirculation

Frequency	Frequency	Percent	Plant	Support	Event	Failed
Initiator	Sequence	Coremelt	Damage	State	Trees	Nodes

PLANT SPECIFIC VALUES

Sequence 1 is a small LOCA with failure of high pressure recirculation. Support state x indicates that two trains of high pressure injection, low pressure injection and containment sprays are available. All auxiliary feedwater pumps are also available. High pressure injection and containment spray injection are successful, the high pressure recirculation has failed, but containment spray recirculation is successful. High pressure recirculation can fail because either the sump valves did not open, the operator did not realign RHR valves to the containment sump. With auxiliary feedwater available, core cooling may be maintained through reflux cooling between the core and steam generators for this sequence.

Sequence 2: Loss of Offsite Power - RCP Seal LOCA

Frequency	Frequency	Percent	Plant	Support	Event	Failed
Initiator	Sequence	Coremelt	Damage	State	Trees	Nodes

PLANT SPECIFIC VALUES

Sequence 2 is an RCP pump seal LOCA during a loss of offsite power. Support state x indicates that there is either no power to train A or no cooling to train B of the safety systems. Hence there are no trains of high pressure, low pressure, RHR or containment spray available. Following the x hours after the pump seal LOCA, if power is restored, then only containment spray injection and recirculation is credited. With auxiliary feedwater available, core cooling may be maintained through reflux cooling between the core and steam generators for this sequence.

TABLE 3-2 DOMINANT ACCIDENT SEQUENCE DISCUSSION
(continued)

Sequence 3 Loss of a Vital AC Power - Failure of Auxiliary Feedwater and Feed and Bleed.

Frequency Initiator	Frequency Sequence	Percent Coremelt	Plant Damage	Support State	Event Trees	Failed Nodes
		PLANT SPECIFIC VALUES				

Sequence 3 is a special initiator: loss of a vital AC power. This will cause a fee water transient and loss of one motor driven auxiliary feedwat r pump. Support state x indicates two trains of high pressure inj ction, low pressure injection, RHR, and containment sprays are initially available. However, only one motor driven auxiliary feedwater pump and the turbine driven auxiliary feedwater pump are available. In this sequence the motor driven and turbine driven auxiliary feedwater pumps fail, the operator initiates the feed and bleed operation but either the operator fails the action or both PORVs do not open. Conta nment spray injection and recirculation are successful.

Sequence 4 Transient - Main & Auxiliary Feedwater and Feed & Bleed Cooling Fail.

Frequency Initiator	Frequency Sequence	Percent Coremelt	Plant Damage	Support State	Event Trees	Failed Nodes
		PLANT SPECIFIC VALUES				

This sequence is a loss of main feedwater transient. Support state x indicates that two trains of high pressure injection, low pressure injection and containment sprays are available. All auxiliary Feedwater pumps are available. Feed and bleed is available as both pressurizer PORVs are available. Both motor driven pumps and the turbine driven auxiliary feedwater pumps fail, main feedwater is not restored and the operator initiates the feed and bleed operation but either the operator fails the action or both PORVs do not open. Containment spray injection and recirculation are successful.

TABLE 3.2-3

Initiating Event	Description	Core Melt Frequency	Percentage of Total
1	Large LOCA		
2	Medium LOCA		
3	Small LOCA		
4	Transient With Main Feedwater		
5	Transient Without Main Feedwater		
6	Loss of Offsite Power		
7	Interfacing Systems LOCA - V Sequence		
8	Loss of One Service Water Train		
9	Loss of One Vital DC Bus 1 or 2		
10	Loss of Both Vital DC Buses		
11	Loss of Vital AC Bus 1 or 2		
12	Loss of Vital AC Bus 3 or 4		

TABLE 3.2-4

TOTAL CORE MELT FREQUENCY BY PLANT DAMAGE STATE

Plant Damage State	Description	Frequency / Reactor Year	Percent Contribution
AEFC	Large LOCA, Early Melt		
AEC	Large LOCA, Early Melt, Failure of Containment Fan Coolers		
AEF	Large LOCA, Early Melt, Failure of Containment Spray		
AE	Large LOCA, Early Melt, No Containment Safeguards		
ALFC	Large 10CA. Late Melt		
ALC	Large LOCA, Late Melt, Failure of Containment Fan Coolers		
ALF	Large LOCA, Late Melt, Failure of Containment Spray		
AL	Large LOCA, Late Melt, No Containment Safeguards		
SEFC	Small LOCA, Early Melt		
SEC	Small LOCA, Early Melt, Failure of Containment Fan Coolers		
SEF	Small LOCA, Early Melt, Failure of Containment Spray		
SE	Small LOCA, Early Melt, No Containment Safeguards		
SLFC	Small LOCA, Late Melt		
SLF	Small LOCA, Late Melt, Failure of Containment Spray		
SLC	Small LOCA, Late Melt, Failure of Containment Fan Coolers		
SL	Small LOCA, Late Melt, No Containment Safeguards		
TEFC	Transient, Early Melt		
TEF	Transient, Early Melt, Failure of Containment Spray		
TEC	Transient, Early Melt, Failure of Containment Fan Coolers		
TE	Transient, Early Melt, No Containment		
V	Interfacing Systems LOCA		
	TOTAL		

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VOLUME 1

ASSESSMENT OF SEVERE ACCIDENT PREVENTION AND MITIGATION FEATURES:

BWR, MARK I CONTAINMENT DESIGN

Prepared by

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Date Published - November 1987

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ABSTRACT

Guidelines and criteria have been developed for preventing and mitigating severe accidents in a BWR which has a Mark I containment (BWR Mark I). The guidelines were developed from insights derived from reviews of in-depth risk assessments performed specifically for the Peach Bottom plant and from assessment of other relevant studies. Accident sequences that dominate the core-damage frequency and those accident sequences that are of potentially high consequence were identified. Vulnerabilities of the BWR Mark I to severe accident containment loads were also identified. In addition, those features of a BWR Mark I, which are important for preventing core damage and are available for mitigating fission-product release to the environment were also identified. These guidelines and criteria are issued to provide direction to an analyst examining an individual plant. This direction calls attention to plant features and operator actions and provides the standards for assessing those features and actions found to be helpful in reducing the overall risk for Peach Bottom and other Mark I plants. Thus, the guidance is offered as a resource in examining the subject plant to determine if the same, or similar, guidelines will be of value in reducing overall plant risk. These guidelines and criteria are intended to serve solely as guidance.

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ACKNOWLEDGMENTS

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NOMENCLATURE

ac	alternating current
A	large loss of coolant accident (LOCA)
ADS	automatic depressurization system
ARC	alternate room cooling
ARI	alternate rod insertion
ASEP	Accident Sequence Evaluation Program
ATWS	anticipated transients without scram
BF	Browns Ferry Nuclear Station
BNL	Brookhaven National Laboratory
BWR	boiling water reactor
C	failure of reactor protection system (RPS)
C_1	mechanical failure to scram
C_2	operator failure to actuate standby liquid control system (SLCS) or to control level with high pressure system (HPS), or failure of SLCS
CDEP	failure of manual depressurization
CDF	core-damage frequency
CHR	containment heat removal
CLWG	Containment Loads Working Group
CONT	inadequate or no containment heat removal leading to loss of core cooling
CPWG	Containment Performance Working Group
CRD	control rod drive system
dc	direct current
DG	diesel generator
DGCM	diesel generators common mode failure
DGREC	failure to recover diesel generators
DHR	decay heat removal
E	failure of coolant injection
E	failure of injection after venting (used for ATWS sequences only)
ECC	emergency core cooling
ECCS	emergency core cooling systems
EPG	Emergency Procedure Guidelines
ESW	emergency service water
FW	feedwater system
GG	Grand Gulf Nuclear Station
GI	generic issue
HADS	failure to inhibit ADS
HEP	human error probability
HPCI	high-pressure coolant injection system
HPIS	high-pressure injection systems
HPLC	failure to control RPV water level with HPCI (either due to operator error and/or hardware failure or malfunction)
HPSW	high pressure service water
I	failure of containment heat removal
IDCOR	Industry Degraded Core Rulemaking Program
INJ	failure of injection with low pressure systems (LPS) after containment failure (CF)
IORV	inadvertent open relief valve
IREP	Interim Reliability Evaluation Program
IPE	individual plant examination
ISL	interfacing system LOCA
J	failure of the HPSW

LLRT	local leak rate testing
LOCA	loss-of-coolant accident
LOOP	loss of offsite power (sometimes denoted by LOSP)
LPCI	low-pressure coolant injection
LPIS	low pressure injection systems
LPLC	failure to control RPV water level at low pressure (either due to operator error and/or hardware failure or malfunction)
LWR	ligh water reactor
MCC	motor control center
MSIV	main steam isolation valve
NPSH	net positive suction lead
NRC	U.S. Nuclear Regulatory Commission
NRC/RES	U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research
P	one or more stuck open relief valves (SORV)
PCS	power conversion system
PRA	probabilistic risk assessment
PWR	pressurized water reactor
Q	failure of feedwater system
Q_1	PCS recovered early
Q_2	PCS recovered late
RB	reactor building
RCIC	reactor core isolation cooling system
RHR	residual heat removal system
RPS	reactor protection system
RPV	reactor pressure vessel
RSS	Reactor Safety Study
RSSMAP	Reactor Safety Study Methodology Application Program
S_1	intermediate LOCA
S_2	small LOCA
SARP	Severe Accident Research Program
SARRP	Severe Accident Risk Reduction Program
SBO	station blackout
SI	safety injection
SLC	failure of SLCS (due to failure of manual initiation and/or due to hardware malfunction or failure)
SLCS	standby liquid control system
SNL	Sandia National Laboratories
SNPS	Shoreham Nuclear Power Station
SORV	stuck open safety relief valve
STCP	Source Term Code Package
SW	service water
(?)	unidentified contribution
(*)	transient sequences are denoted by T followed by letters denoting the relevant failure, e.g., TC transients involving failure of RPS TQUV transients involving failure of FW, HPIS, and LPIS, etc.
T	transient
TAF	top of active fuel
TB	station blackout sequence (also referred to as SBO)
TC	ATWS
TCV	loss of condenser vacuum initiator

NOMENCLATURE (Cont'd)

T_{FW}	loss of feedwater initiator
T_I	isolation transients
$T_I^{>25\%}$	isolation transients at power level greater than 25%.
T_M	manual shutdown
T_{NOPCS}	transients with PCS initially unavailable
T_{PCS}	transients with PCS initially available
TPQI	transient sequence with SORV failure, feedwater failure and loss of CHR
TQU	transient sequence in which the feedwater and HPI systems fail
TQUX	transient sequence in which the feedwater and HPI systems fail and depressurization does not occur
T_T	turbine trip initiator
$T_T^{>25\%}$	turbine trip transients at power level greater than 25%.
TW	loss of CHR
TC	ATWS
U	failure of high pressure injection system (HPIS)
USI	unresolved safety issue
V	failure of low pressure injection system (LPIS)
VCS	operator failure to control level and reactivity with low pressure systems
VENT	containment (wetwell) venting failure
W	failure of containment heat removal (CHR)
X	failure of reactor pressure vessel (RPV) depressurization

1. EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) has formulated an approach for a systematic safety examination of existing plants to determine whether particular severe accident vulnerabilities are present and what changes are desirable to ensure that there is no undue risk to public health and safety.

The Industry Degraded Core Rulemaking Program (IDCOR) selected four reference plants for detailed analysis: Peach Bottom, Grand Gulf, Sequoyah, and Zion. The IDCOR analyses performed for the reference plants have been documented together with the methodology used for the analyses and the technical basis supporting the methodology.

Parallel with the IDCOR work, the NRC under the Severe Accident Research Program (SARP), performed risk assessments, audit calculations, sensitivity studies, and uncertainty analyses for five plants. The five plants considered by SARP were Peach Bottom, Grand Gulf, Sequoyah, Zion and Surry.

The purpose of this effort is to review all of the IDCOR and SARP analyses performed for the reference plants, understand the reasons for the differences, and then use the experience gained from these reviews for developing guidelines and criteria that identify plant features and operator actions that were found to be important for either preventing or mitigating severe accidents in each plant type. In turn, these guidelines should prove helpful in the systematic safety examination of individual plants.

Three basic objectives or goals for this severe accident program apply equally to all plant types:

- Goal 1: Mitigate fission-product releases.
- Goal 2: Control the frequency of high-consequence sequences.
- Goal 3: Reduce high core-damage frequency.

The aim was, therefore, to develop detailed guidelines and criteria that could be used to achieve these goals during the examination of individual plants.

"Guidelines," as used in this report, identify those plant features and operator actions that were found to be important to either preventing or mitigating severe accidents in the reference plant studies. These guidelines provide a list of plant features and operator actions that the utilities can use as part of each individual plant examination (IPE). It is not the intent of this report to specify a set of improvements for either the reference plant or for any other plant which would be sufficient to achieve a certain level of safety. Instead, the guidelines indicate potential improvements in various areas of plant design and operation of which each utility should be aware when conducting its IPE and making decisions on plant improvements. The intent is to provide guidance to the analyst performing an IPE, as to the plant features and operator actions which were found to reduce overall risk. It is prudent to check whether potential improvements identified in studies of other similar plants can be of help in improving overall plant performance. The guidelines contained in this report, therefore, complement the IPEs.

"Criteria," as used in this report, are the attributes which have been identified as important to assess the performance of plant features and operator actions identified in the guidelines. The criteria provide deterministic (as opposed to probabilistic) performance measures which are judged to be helpful to implement the concepts contained in the guidelines. When a decision is made to provide an item or an action specified in a guideline, the utility should address a set of questions relating to the design, operation and availability of the needed equipment and the training of operators. For the example of containment-venting guidance, the capacity of the venting system, the selection of setpoints to initiate venting, the availability of applicable procedures and the accessibility of certain valves by operators should be addressed. The criteria on containment venting provide helpful information in assessing venting capability in each individual plant.

Based on an extensive review of prior severe accident investigations, the authors have provided a set of guidelines and associated criteria which can be used to assess the capability of individual boiling water reactor (BWR), Mark I plants to cope with severe accidents. Although much of the work is based on probabilistic risk assessments (PRAs), the guidelines and criteria are deterministic in nature. That is the criteria describe specific features of key systems and operational procedures which have been found helpful in reducing the likelihood of severe accidents. The guidelines and criteria take into account detailed severe accident experiments and analyses performed by the NRC/RES, the nuclear power industry and foreign governments.

The following sections present the insights gained from reviewing the PRAs. Specifically, the IDCOR Peach Bottom Integrated Containment Analyses[1] and the SARP Peach Bottom reports[2-4] were reviewed in detail. These studies were compared with the original Peach Bottom risk assessment in the Reactor Safety Study (RSS) (WASH-1400)[5] and relevant BWR PRAs for other plants, namely Browns Ferry,[6] Limerick[7,8] and Shoreham.[9,10]

1.1 Core-Damage Profile

PRAs for BWRs have indicated that accidents initiated by transients rather than loss-of-coolant accidents (LOCAs) dominated the total core-damage frequency (CDF) estimates. However, there appeared to be no consistent pattern of relative ranking of transient sequences among the PRAs reviewed. It is also important to observe that for a given accident sequence, the major contributor to differences in quantitative results between the PRAs was subjective modeling assumptions rather than plant differences or data differences. For the four BWR plants considered in the six PRAs that were examined, the same few functional accident sequences figured prominently in all of the respective CDF profiles.

In the RSS[5] (which used the Peach Bottom plant) and the Interim Reliability Evaluation Program (IREP) study[6] (which used the Browns Ferry plant) loss of containment heat removal sequences were found to be important contributors to core melt (about 50%). The more recent Accident Sequence Evaluation Program (ASEP) and IDCOR studies have reduced the CDF, attributable to these sequences, based on operating procedures that include venting and alternative injection.

For the Limerick PRA, Browns Ferry IREP and Shoreham PRA, accident sequences with failure of high-pressure injection were important contributors to CDF. Most of this contribution was because of a high failure rate for the automatic depressurization system (ADS).

Both SARP and IDCOR indicated that station blackout (SBO) and anticipated transients without scram (ATWS) are the dominant core-melt sequences for the Peach Bottom plant. Both studies calculated a total CDF approaching 10^{-5} per reactor year for these events.

1.2 Consequence Analysis

The assessment of core-meltdown phenomena and containment response in the PRAs indicated that the Mark I containment is vulnerable to severe accident containment loads. Unless mitigative actions are taken, a Mark I containment has the potential to fail shortly (in a few hours) after the core debris melts through the reactor pressure vessel. If containment failure occurs in the drywell, any fission products in the drywell atmosphere could pass to the reactor building and ultimately to the environment without the benefit of suppression pool scrubbing. Because of this vulnerability, the predicted offsite consequences were relatively insensitive to the definition of the accident sequence. In addition, differences in the IDCOR and Severe Accident Risk Reduction Program (SARRP) assessments of containment response and fission-product release also did not result in major differences in the predicted offsite consequences. The only time that a major reduction in offsite consequences was predicted by IDCOR and SARRP was with successful wetwell venting and no suppression pool bypass.

SARRP estimated that the dominant accident sequences (namely ATWS and SBO) result in a significant probability of suppression pool bypass. Thus, the bypass mechanisms identified in the SARRP analysis have to be addressed to ensure mitigating of fission products.

1.3 Guidelines and Criteria

Each guideline is provided with a detailed test of criteria which provide helpful information to assess the performance of plant features and operator actions identified in the guidelines.

1.3.1 Mitigate Fission-Product Releases

The assessment of core-meltdown phenomena and containment response indicates that the Mark I containment is vulnerable to severe accident containment loads because of its relatively small volume. Unless mitigative actions are taken, a Mark I containment has the potential to fail shortly after the core debris melts through the reactor pressure vessel. For this reason the authors developed Guidelines 1 and 2, as shown in Table 1.1, which have the potential to mitigate the consequences of a severe accident.

Guideline 1 - Maintain Containment Integrity

Mark I containments are very effective at condensing steam, but their small volume makes them vulnerable to any combustible and noncondensable gases that would be generated during a severe core-meltdown accident. The impact of

hydrogen burning was not significant for the Mark I containment because the atmosphere is inerted. However, SARRP predicts that the accumulation, of noncondensable gases, released from core-concrete interaction, will fail the containment because of overpressurization. IDCOR, on the other hand, had predicted that the containment would fail because of high temperatures before overpressurization. Differences in the predicted drywell pressure and temperature histories will influence the containment performance. In order to address this uncertainty in the failure mode, the authors developed Guideline 1.A (see Table 1.1) on containment venting (relating to preventing overpressure failure) and Guideline 2.B. on containment spray (relating to preventing of failure because of high temperature).

Guideline 2 - Maintain Suppression Pool Effectiveness

The ability of the Mark I suppression pool to trap aerosol fission products is an important mitigative feature since it leads to a direct reduction in offsite consequences by a factor of 10 or more. Thus, any pathways that might open, which would allow the fission products to bypass the pool, are undesirable. Therefore, Guideline 2 is intended to prevent fission-products from bypassing the suppression pool.

1.3.2 Control the Frequency of High-Consequence Sequences

Guideline 3 - Interfacing Systems LOCA

In general, BWR Mark I PRAs have found interfacing systems LOCA to be extremely unlikely. However, the possibility of high releases makes it important to ensure that the frequency of these events is kept very low at all Mark I plants.

Guideline 4 - Anticipated Transients Without Scram

Anticipated transients without scram (ATWS) have been found to be important contributors to risk for many BWRs. The NRC promulgated the ATWS rule to reduce the frequency of ATWS events. The criteria developed for this guideline emphasize human reliability insights as to the importance of correct emergency procedures and operator training in recovering from an ATWS event.

1.3.3 Reduce High Core-Damage Frequency Sequences

Guideline 5 - Station Blackout

For accidents involving the loss of offsite power and onsite emergency power, the NRC recommends examining the proposed station blackout (SBO) rule for applicability. The criteria associated with Guideline 5 are intended to emphasize the need to search for plant specific features and potential common cause failures which could disable systems required to work during an SBO. For individual plants which are found to have a vulnerability to SBO, the criteria highlight the importance of proper emergency procedures and operator training in recovering from an SBO event.

Guideline 6 - Loss of Containment Heat Removal

Accident sequences involving loss of containment heat removal (CHR) were found to be quite important in the earlier PRA studies reviewed. In WASH-1400 loss of CHR sequences accounted for 53% of the calculated CDF. In the Browns Ferry IREP study, these sequences similarly accounted for 50% of the calculated core damage frequency. However, the most recent Peach Bottom studies, IDCOR and ASEP/SARRP, show a two-and-three-order-of-magnitude reduction, respectively, in the quantification of these sequences. Therefore, Guideline 6 has been developed to generically address the mechanisms already effectively employed at Peach Bottom to reduce the frequency of loss of CHR sequences.

Guideline 7 - Reactor Pressure Vessel (RPV) Depressurization Performance

One of the insights gained from the existing BWR PRAs is the importance of the ADS in mitigating loss of high pressure injection sequences. Specific criteria have been developed to ensure that the likelihood of these accident sequences occurring is low.

Guideline 8 - Support System Interdependencies

Although the importance is difficult to quantify, one of the insights developed in most risk assessment studies is the importance of support system interdependencies. For example, a draft of the SARRP Peach Bottom study indicated that loss of all service water was a dominant contributor to core melt. The final version of the accident sequence studies[2] has reduced it to one percent of the overall core melt frequency. In order to ensure that support system vulnerabilities do not cause unacceptably high core melt frequencies for other BWR Mark I plants, the authors have developed this guideline to help assess any weaknesses of the support systems.

Guideline 9 - Flooding Within Reactor Building

Flooding of the reactor building has been found to be a significant contributor to core damage in only one Mark II plant. However, the concerns appear to be of general applicability to other designs. Thus, this guideline was developed for Mark I plants to assess the potential for flooding of safety-related equipment.

1.4 Using the Guidelines and Criteria

Numerous investigations, including PRAs, have been performed for the reference plants and similar plants by both the NRC and the nuclear power industry. The insights gained from many of the studies have been used in developing the guidelines and criteria contained in this report (including the other volumes of this report). These guidelines and criteria are issued to provide guidance to the analyst performing an IPE. This guidance is in the form of plant features, operator actions and the criteria for assessing those features and actions found to be helpful in reducing the overall risk for Peach Bottom and other Mark I plants. Thus, the guidance is given to provide a resource in examining the subject plant to determine if the same, or similar, guidelines will be of value in reducing overall plant risk. These guidelines and criteria are intended to be used solely as guidance.

1.5 References for Section 1

1. "Peach Bottom Atomic Power Station-Integrated Containment Analyses," IDCOR Technical Report T23.1PB, March 1985.

2. A. M. Kolaczkowski et al., "Analysis of Core Damage Frequency from Internal Events: Peach Bottom, Unit 2," Sandia National Laboratories, NUREG/CR-4550, Volume 4, October 1986.

3. C. N. Amos et al., "Containment Event Analysis for Postulated Severe Accidents: Peach Bottom Atomic Power Station, Unit 2," Sandia National Laboratories, NUREG/CR-4700, Vol. 3, Draft Report for Comment, May 1987.

4. C. N. Amos et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Peach Bottom, Unit 2," Sandia National Laboratories, NUREG/CR-4551, Vol. 3, Draft for Comment, April 1987.

5. "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, NUREG/75-014, October 1975.

6. S. E. Mays et al., "Interim Reliability Evaluation Program: Analysis of the Browns Ferry, Unit 1, Nuclear Plant," Idaho National Engineering Laboratory, NUREG/CR-2802, July 1982.

7. "Probabilistic Risk Assessment - Limerick Generating Station," Philadelphia Electric Co., 1982.

8. I. A. Papazoglou et al., "A Review of the Limerick Generating Station Probabilistic Risk Assessment," Brookhaven National Laboratory, NUREG/CR-3028, February 1983.

9. "Probabilistic Risk Assessment - Shoreham Nuclear Power Station," Long Island Lighting Company, June 1983.

10. D. Ilberg et al., "A Review of the Shoreham Nuclear Power Station Probabilistic Risk Assessment (Internal Events and Core Damage Frequency)," Brookhaven National Laboratory, NUREG/CR-4050, June 1985.

Table 1.1 Guidelines for Preventing and Mitigating Severe Accidents in a BWR with a Mark I Containment

Guideline	Description
	Mitigate Fission-Product Releases:
1	Maintain Containment Integrity
1.A.	Provide Wetwell Venting
2	Maintain Suppression Pool Effectiveness
2.A.	Prevent Suppression Pool Bypass
2.B.	Provide Drywell Spray
	Control the Frequency of High-Consequence Sequences:
3	Interfacing Systems LOCA
3.A.	Prevent Overpressurization of Low Pressure Systems
4	Anticipated Transients Without Scram (ATWS)
4.A.	Provide Operator Response During ATWS
	Reduce High Core-Damage Frequency Sequences:
5	Station Blackout
5.A.	Provide Reactor Pressure Vessel Injection
6	Loss of Containment Heat Removal
6.A.	Provide Long-Term Emergency Core Cooling
7	Reactor Pressure Vessel (RPV) Depressurization Performance
7.A.	Provide RPV Depressurization
8	Support System Interdependencies
8.A.	Examine Support System Interdependencies
9	Flooding Within Reactor Building
9.A.	Prevent and Mitigate Reactor Building Flooding

2. INTRODUCTION

2.1 Background

The U.S. Nuclear Regulatory Commission (NRC) has formulated an approach for a systematic safety examination of existing plants to determine whether particular severe accident vulnerabilities are present and what changes are desirable to ensure that there is no undue risk to public health and safety.

The Industry Degraded Core Rulemaking Program (IDCOR) selected four reference plants for detailed analysis, namely:

- Peach Bottom (a BWR with a Mark I containment)
- Grand Gulf (a BWR with a Mark III containment)
- Zion (a PWR with a large dry containment)
- Sequoyah (a PWR with an ice condenser containment)

The IDCOR analyses performed for the above reference plants have been documented together with the methodology used for the analyses and the technical basis supporting the methodology.

Parallel with the IDCOR work, the NRC under the Severe Accident Research Program (SARP), performed risk assessments, audit calculations, sensitivity studies, and uncertainty analyses for five plants. The five plants considered include the above four IDCOR reference plants, and, in addition

- Surry (a PWR with a subatmospheric containment)

The purpose of this effort is to review all of the IDCOR and SARP analyses performed for the reference plants, understand the reasons for the differences, and then use the experience gained from these reviews for developing guidelines and criteria that identify plant features and operator actions found to be important for either preventing or mitigating severe accidents in each plant type. In turn, these guidelines should be helpful in the systematic safety examination of individual plants.

The first plant reviewed was Peach Bottom, which is a BWR-4 with a Mark I containment. The IDCOR Peach Bottom analysis[1] was documented in March 1985 and was supplemented by additional sensitivity studies in July 1985. The SARP Peach Bottom reports[2-4] were reviewed in draft form during 1986. These reports were published early in 1987 and were summarized in the "Reactor Risk Reference Document" (NUREG-1150),[5] which was published for comment in February 1987. The experience gained from the review of these Peach Bottom studies along with other BWR PRA studies (namely, Limerick, Shoreham and Browns Ferry) was used to generate the guidelines and criteria which are the subject of this report.

2.2 Objectives

Three basic objectives or goals for this severe accident program apply equally to all plant types:

- Goal 1: Mitigate fission-product releases.
- Goal 2: Control the frequency of high-consequence sequences.

- Goal 3: Reduce high core-damage frequency.

The aim was, therefore, to develop detailed guidelines and criteria that could be used to achieve these goals during the examination of individual plants. The guidelines and criteria are defined in the sections that follow.

2.2.1 Guidelines

"Guidelines," as used in this report, identify those plant features and operator actions that were found to be important to either preventing or mitigating severe accidents in the reference plant studies. These guidelines provide a list of plant features and operator actions that the utilities can use as part of each individual plant examination (IPE). It is not the intent of this report to specify a set of improvements for either the reference plant or for any other plant which would be sufficient to achieve a certain level of safety. Instead, the guidelines indicate potential improvements in various areas of plant design and operation of which each utility should be aware when conducting its IPE and making decisions on plant improvements. The intent is to provide guidance to the analyst performing an IPE, as to the plant features and operator actions which were found to reduce overall risk. It is prudent to check whether potential improvements identified in studies of other similar plants can be of help in improving overall plant performance. The guidelines contained in this report, therefore, complement the IPEs.

The three objectives or goals were noted as applying equally to all plant types. Although the goals are independent of plant type, the guidelines that are needed to achieve the goals are plant dependent. In general terms, Goal 1 implies that there should be effective means of mitigating the fission-product releases for the broad classes of accident sequences which dominate the core-damage frequency. Therefore, these dominant accident sequences have to be determined and those plant features and operator actions that are available to mitigate the release of fission products have to be identified. Only then can detailed guidelines be developed to ensure that these dominant accident sequences can be mitigated.

There may be accident sequences for which a specific plant will have substantial fission-product releases (e.g., containment bypass sequences). Thus, for such sequences Goal 1 may be difficult to achieve. Therefore, all reasonable steps are identified which could reduce the frequency of these potentially high-consequence sequences (namely Goal 2). Again, the accident sequences have to be identified and plant vulnerabilities and/or operator actions that lead to core damage for these sequences also have to be identified. Detailed guidelines can then be developed which will aid in assessing an individual plant's capability to prevent these sequences from occurring.

It is also desirable to ensure that the overall core-damage frequency is low (namely Goal 3). Again, the dominant accident sequences have to be found so that detailed guidelines can be developed to reduce the frequency of these sequences, if necessary.

In general, the following screening process was used to determine whether or not to develop a particular guideline:

- any accident sequence with a core-damage frequency greater than 10^{-6} per reactor year
- any sequence that contributed to more than 5% of the total core-damage frequency
- any event that caused a conditional probability of early containment failure greater than 0.1
- any sequence that resulted in containment bypass with a frequency greater than 10^{-7} per reactor year
- any sequence that was judged to be uniquely important (example, very severe consequences)

This screening process led to the development of guidelines that can be used in the systematic safety examination of other BWRs with Mark I containments. For example, the guideline for venting of the wetwell was identified as an item that would help to achieve Goal 1 (namely, to mitigate fission-product releases) for the BWR Mark I reference plant. Therefore, in the safety examination of other BWRs with Mark I containments, the need for wetwell venting may need to be carefully assessed.

The development of a particular guideline for the BWR Mark I reference plant does not imply that this plant or any of the other plants in this category need to conform to this guideline. It simply means that analyses have indicated that this particular guideline has the potential to significantly reduce risk. Thus, the guidance is given to provide a resource in examining the subject plant to determine whether the same or similar guidelines will be of value in reducing overall plant risk. Whether or not the guideline is useful or needed in a particular BWR with a Mark I containment depends on plant-specific details and is beyond the scope of this report and is therefore not addressed here.

2.2.2 Criteria

"Criteria," as used in this report, are the attributes which have been identified as important to assess the performance of plant features and operator actions identified in the guidelines. The criteria provide deterministic (as opposed to probabilistic) performance measures which are judged to be helpful to implement the concepts contained in the guidelines. When a decision is made to provide an item or an action specified in a guideline, the utility should address a set of questions relating to the design, operation and availability of the needed equipment and the training of operators. For the example of containment-venting guidance, the capacity of the venting system, the selection of setpoints to initiate venting, the availability of applicable procedures and the accessibility of certain valves by operators should be addressed. The criteria on containment venting provide helpful information in assessing venting capability in each individual plant.

The criteria address the general issues of (1) survivability of equipment (i.e., whenever credit is given for a system or a component to mitigate the accident, the ability of the equipment to function under environmental and fluid dynamic loads associated with severe-accident sequences must be taken

into account), (2) equipment capabilities, capacities, and duration of operability, (3) accessibility of equipment, (4) availability of support systems, (5) identification of necessary components, (6) identification of important operator actions, and (7) identification of parameters for initiation of mitigating systems and operator actions.

2.3 Organization of the Report

This report describes detailed guidelines and criteria for preventing and mitigating severe accidents in BWRs that have a Mark I containment. It is the first of a series of five volumes that deal with guidelines and criteria for several different reactor and containment types. Other volumes in the series are:

- Volume 2: BWRs with Mark II Containments
- Volume 3: BWRs with Mark III Containments
- Volume 4: PWRs with Large-Volume Containments
- Volume 5: PWRs with Ice Condenser Containments.

Appendix A of this volume contains a review of the IDCOR and SARP analyses for a BWR-4 with a Mark I containment along with other pertinent studies. The insights gained from these studies lead to the identification of the strengths and vulnerabilities of a BWR with a Mark I containment. In Section 3, the three basic goals of the program are related to the relevant design features and operating characteristics of a BWR-4 with a Mark I containment. The guidelines recommended to achieve the three goals are therefore initially developed in Section 3. In Section 4, the guidelines are restated and detailed criteria are developed for each guideline. Appendix B highlights the key systems and features of a BWR Mark I plant which ensure that the severe accident risk is acceptably low.

2.4 References for Section 2

1. "Peach Bottom Atomic Power Station-Integrated Containment Analyses," IDCOR Technical Report T23.1PB, March 1985.

2. A. M. Kolaczkowski et al., "Analysis of Core Damage Frequency from Internal Events: Peach Bottom, Unit 2," Sandia National Laboratories, NUREG/CR-4550, Volume 4, October 1986.

3. C. N. Amos et al., "Containment Event Analysis for Postulated Severe Accidents: Peach Bottom Atomic Power Station, Unit 2," Sandia National Laboratories, NUREG/CR-4700, Vol. 3, Draft Report for Comment, May 1987.

4. C. N. Amos et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Peach Bottom, Unit 2," Sandia National Laboratories, NUREG/CR-4551, Vol. 3, Draft for Comment, April 1987.

5. "Reactor Risk Reference Document," U.S. Nuclear Regulatory Commission, NUREG-1150, Draft for Comment, February 1987.

3. DEFINITION OF GOALS AND RELEVANT BWR MARK I FEATURES

In Section 2 of this report, the concept of three basic objectives or goals for this severe accident program was introduced. The concept applied equally to all plant types. In this section, the three goals are related to the relevant design features and operating characteristics of a BWR with a Mark I containment for the accident sequences and containment failure modes found to be important in Appendix A. This includes consideration of both favorable and unfavorable severe accident attributes.

Screening criteria have been used to identify those sequences that need to be addressed by severe accident guidelines for each goal. Specifically:

- For Goal 1 (Mitigate fission-product releases), all sequences have been examined which represent 5% of the core-melt frequency or are estimated to occur more often than 10^{-6} per reactor year and which result in a conditional probability of early containment failure greater than 0.1.
- For Goal 2 (Control the frequency of high-consequence sequences), all sequences have been examined which result in pool bypass and are estimated to occur more often than 10^{-7} per reactor year.
- For Goal 3 (Reduce high core-damage frequency sequences), all sequences have been examined which "have the potential to occur" more frequently than 10^{-6} per reactor year. Note that this screening criterion has been used to identify potential vulnerabilities from risk assessment insights which do not necessarily apply to Peach Bottom itself, but may apply to other Mark I plants.

This section provides the link between the goals (developed in Section 2) and the guidelines (developed in Section 4) that may be used to assess the capability of specific plants to meet these goals. This section is organized into three subsections, which correspond to the three goals.

3.1 Mitigate Fission-Product Releases

Goal 1 requires that there shall be effective means of mitigating the fission-product releases for the broad classes of accident sequences that may lead to core damage in a BWR with a Mark I containment. In Appendix A the most important contributors to the core-damage frequency (CDF) were found to be SBO and ATWS sequences. Several studies indicate that transients with loss of injection into the reactor pressure vessel (RPV) are also potentially important contributors. Other transients and loss-of-coolant accidents (LOCAs) may also contribute to the CDF. Two specific accident sequences for which mitigation by the Mark I containment is ineffective are also identified in Appendix A. These specific sequences are discussed further in Section 3.2, which attempts to determine how the frequency of these unmitigated sequences can be reduced. This section concentrates on the broad classes of accident sequences for which plant features provide significant means of mitigating fission-product release. In the following sections both the favorable and unfavorable severe accident attributes of the Mark I containment are identified. This discussion in turn leads to the development in Section 3.1.3 of the two guidelines that are related to Goal 1.

3.1.1 Plant Vulnerabilities

The Mark I containment is a small-volume, pressure-suppression design. The suppression pool is available to condense steam released from the primary system during an accident. However, the small volume of the Mark I containment makes it vulnerable to pressure and/or temperature increases because of the noncondensable gases and heat released during a core-meltdown accident. There are differences between the IDCOR[1] and SARP[2] analyses regarding the estimates of how long it will take to pressurize a Mark I containment to its ultimate capacity after the core debris has failed the reactor vessel (and is interacting with concrete); but both studies concluded that the containment will eventually fail. Therefore, unless mitigative actions are taken, a Mark I containment will fail because of overpressure or overtemperature within a few hours of RPV failure. If containment failure occurs in the drywell, any fission products in the drywell atmosphere could pass to the reactor building (and ultimately to the environment) without the benefit of suppression pool scrubbing. Note that suppression pool scrubbing is an important mitigative feature of a Mark I containment (refer to Section 3.1.2).

An inspection of the Mark I containment configuration (see Figure A.1) shows that the pedestal below the RPV would tend to confine the core debris after a core-meltdown accident. Extensive core-concrete interactions would be expected to occur. There are differences between the IDCOR and SARP analyses related to how high the core debris temperature will remain during these interactions and to the quantities of the less volatile fission products that will be released. However, at this time the possibility of the core debris remaining hot and releasing significant quantities of fission products has not been ruled out.

After the region directly underneath the RPV (pedestal region) filled with core debris, there would be sufficient core materials from a full core-meltdown to overflow onto the drywell floor. If the core debris remained molten it could flow across the drywell floor and reach the steel containment liner. This steel liner would offer little resistance to molten core debris and it was predicted[3] to fail very rapidly if the core debris reached the liner wall. This was found to be a mechanism for early loss of drywell integrity in the SARP analysis and is thus another Mark I containment vulnerability relative to some other containment designs in which the geometry would tend to prevent the core debris from reaching the containment wall. The SARP analysis estimated a relatively high conditional probability for liner failure for SBO sequences.

In the sections that follow, suppression pool scrubbing is noted as an effective mitigative feature for the Mark I containment provided all of the fission products pass through the pool. It is, therefore, important to ensure that paths do not open which would allow the fission products to bypass the suppression pool. The vacuum breakers between the wetwell and drywell would create a path that bypasses the suppression pool, if they fail open. In addition, the various drywell penetration seals could be degraded at high temperatures and pressures. Failure of these seals would also open up paths that would bypass the suppression pool. If the main steam isolation valves (MSIVs) fail to close, another suppression pool bypass path would exist.

Although the Mark I containments appear to be vulnerable to severe accident containment, they have several very important mitigative features, which are described in the section that follows.

3.1.2 Mitigating Features

The suppression pool in a Mark I containment is a very effective mechanism for trapping any fission-product aerosols that might pass through it. Thus, to a large extent the suppression pool has the potential to compensate for the vulnerabilities identified above (in Section 3.1.1). For example, overpressure failure of the containment (and perhaps loss of drywell integrity) can be prevented by venting the wetwell. With venting of the wetwell atmosphere, containment integrity is lost but the containment function (retention of the fission products in the pool) is maintained.

High drywell temperatures and resultant penetration seal degradation can be prevented by drywell spray. The potential for molten core debris to spread across the drywell floor and fail the containment liner may also be reduced by spray operation (refer to Section A.2.4). Drywell spray will also contribute to decontamination of the drywell atmosphere even for sequences with substantial suppression pool bypass.

The atmosphere in a Mark I primary containment is continuously inerted (by introducing nitrogen and thereby lowering the oxygen concentration) during operation, which prevents hydrogen combustion. This is a very significant mitigative feature, which is important to maintain during a severe accident. For example, wetwell venting and drywell spray operation could result in a vacuum in the containment, which could introduce additional oxygen and thus deinert the containment atmosphere.

An area of significant phenomenological uncertainty (refer to Section A.2) relates to core meltdown with the primary system at high pressure. If molten core materials are ejected from the RPV under pressure, it has been suggested in the SARP analysis that the materials form fine aerosols, which could be dispersed into the containment atmosphere and directly heat it. This could result in a large pressure pulse, which could threaten containment integrity at the time of RPV failure. BWRs have an automatic depressurization system (ADS) which can prevent high-pressure core-meltdown (on RPV low water level after a time delay or manually by the operator). The ADS has a dual role as a core-melt prevention system as well. For those sequences in which the high-pressure injection systems fail (TQU), the ADS can depressurize the reactor vessel and allow the low-pressure systems to inject water into the RPV.

Finally, the BWR Mark I primary containment is completely enclosed in a reactor building. This building is, therefore, available as a secondary containment to trap any fission products that might be released from the primary containment during a severe accident. The amount of fission products that might be trapped in the reactor building is uncertain, but it is a potentially important mitigating feature.

3.1.3 Maintain Containment Integrity and Suppression Pool Effectiveness

The above discussion has identified several plant features of the BWR plant with a Mark I containment that have the potential to help achieve Goal 1, namely, mitigating fission-product releases. From the above discussion, two guidelines have been developed and related to these features that will aid in assessing whether specific plants meet Goal 1. The guidelines address containment integrity, the effectiveness of the suppression pool, and the various mechanisms for possible pool bypass. As long as the dominant release path is through the suppression pool, the consequences of core melt accidents were shown (refer to Table A.12) to be reduced by at least a factor of 10 relative to sequences that bypassed the pool. Guideline 2 also addresses one of the dominant pool bypass sequences (e.g., melt-through of the steel containment) to ensure that the release is substantially reduced or that the frequency is kept low.

Both the ASEP and IDCOR studies identified SBO and ATWS sequences as being the most significant contributors to core melt. Although the studies disagree on which is more important, they are in general agreement on the total core-melt probability (about 10^{-5}/reactor year). Thus, it is believed that the effectiveness of the suppression pool must be maintained for both ATWS and SBO events.

For sequences that threaten the containment by overpressure, wetwell venting has the potential to preserve the containment function by relieving noncondensable gases and/or saturated steam, thus preventing further pressure buildup while forcing fission products to be scrubbed by the pool. However, for the two dominant sequences (SBO and ATWS), existing venting procedures will be difficult to perform. For SBO sequences, power dependencies may preclude actuation of venting from the control room, and high radiation levels may hamper local manual valve actuation. For ATWS sequences, the large venting capacity requirements, short time frame for operator action and possible problems with normal isolation systems make successful venting under such conditions operationally difficult. Detailed criteria are developed for this guideline in Section 4.1.1 to ensure venting capability for these dominant accident sequences, thus maintaining suppression pool effectiveness.

The SARRP event trees for Peach Bottom indicated that core debris melting through the steel containment shell was a dominant suppression pool bypass mechanism. SARRP estimated that the conditional probability of containment shell melt-through was relatively high for SBO sequences. Although it could be argued that, even with containment shell melt-through, if the drywell pressure is relieved by wetwell venting, the driving force to transport the radionuclide from the drywell to the reactor building will be reduced, the fractional fission product release remains uncertain. Therefore it appears prudent to attempt to keep the containment shell from melting. Ceramic brick curbs may be built to keep the corium from reaching the containment shell but complete debris confinement would be difficult without impacting water return to the pool. If both overpressure failure and containment melt-through can be prevented (by venting and by building corium retainers) the likelihood of pool bypass can be reduced substantially (by about a factor of 10 using SARP event trees).

For sequences that still result in suppression pool bypass, the drywell sprays will tend to wash out aerosols from the containment atmosphere and thus reduce the airborne fission product concentration during core-concrete interaction. In some sequences such as SBO, drywell and wetwell sprays would not be available because of ac power requirements. For the other dominant sequence class (ATWS), these sprays may not be available because of suppression pool heatup and its effects on net positive suction head (NPSH) of the spray pumps. The guidelines and criteria developed in Section 4.1.1 address alternative power supplies and suction sources to ensure that drywell sprays will be available for the two dominant sequences.

3.2 Control the Frequency of High-Consequence Sequences

The plant features identified in Section 3.1 can effectively mitigate fission-product releases for the broad classes of accident sequences that were found to dominate the core-damage frequency (CDF). However, two accident sequences were identified in Appendix A for which substantial reducing of fission-product release for the BWR Mark I plant cannot be ensured. Neither of these two sequences appear to meet the screening criteria ($>10^{-7}$/reactor year) for Peach Bottom, but their importance in other PRAs indicate that guidance should be developed to ensure that specific plant vulnerabilities do not make them contributors to risk for other BWR Mark I plants.

The first accident sequence that may defeat the plant containment features identified in Section 3.1 is the interfacing LOCA sequence. Although none of the BWR Mark I PRAs reviewed in Appendix A indicate that it is a significant contributor to core-melt frequency, PRAs for othe plant types (e.g., Shoreham) have identified it as a significant contributor to risk. It has also been identified as a generic issue (GI-105) by the NRC. Thus, Guideline 3 and associated criteria are developed in Section 4.2.1 to ensure that other BWR Mark I plants review the potential contribution of interfacing systems LOCA to risk. This guideline and criteria should be considered appropriate pending resolution of GI-105.

The second accident sequence that may defeat several of the plant features identified in Section 3.1 is an ATWS with a power transient. In this sequence, the operator fails to control the RPV injection at low pressure during an ATWS event. The rising water level in the reactor vessel produces a power transient that cannot be controlled by the normal containment heat removal systems. The containment will pressurize rapidly and may fail with the resultant loss of coolant injection and eventual core melt into a failed containment. The ability to control this rapidly progressing sequence by wetwell venting is difficult, and thus mitigating this sequence appears to be unlikely. Therefore, the risk of this subset of the ATWS sequences must be controlled by ensuring that its frequency is low. In Section 4.2.2, a guideline and detailed criteria are developed related to operator actions during an ATWS event.

3.3 Reduce High Core-Damage Frequency Sequences

In Section A.1 it was found that only a few accident sequences figure prominently in the core-damage profiles of all of the PRAs reviewed. This led to the conclusion that if the frequency of this relatively small subset of accident sequences could be reduced, then the overall CDF could also be reduced.

3.3.1 Station Blackout

Most of the PRAs for BWRs (including the ASEP study of Peach Bottom) indicated that the most important contributor to the CDF is station blackout (SBO). Therefore, a severe accident guideline with specific criteria has been developed in Section 4 related to these accident sequences.

Station blackout refers to a loss of the offsite power supply with concurrent failure of the two emergency ac power divisions. Reducing SBO sequences is addressed by the proposed NRC SBO rule. The guideline and associated criteria developed by the present study emphasize the need to search for plant specific features and potential common cause failures which could disable systems required to work during an SBO. For individual plants which are found to have a vulnerability to SBO, the criteria highlight the importance of proper emergency procedures and operator training in recovering from an SBO event.

For the BWR-4 design, the two systems designed to operate in the presence of an SBO are the high pressure coolant injection (HPCI) and the reactor core isolation cooling (RCIC) systems. By removing the long-term SBO sequences related to dependent failure modes of either system, the SBO core-damage frequency can be significantly reduced. The long-term dependent failure modes of HPCI and RCIC under SBO conditions are (1) battery depletion, (2) pump seal failure because of suppression pool heat-up, and (3) loss of room cooling. Since the RSS[4] did not address these failure modes, it is expected that their removal from the current design would lower the CDF because of blackout back to essentially the RSS point estimate of 10^{-7}.

3.3.2 Loss of Containment Heat Removal

Accident sequences involving loss of containment heat removal (CHR) (e.g., TW) were found to be important in the earlier PRA studies considered in Appendix A (see Table A.1). In the RSS,[4] the TW sequences accounted for 53% of the calculated CDF. In the Browns Ferry IREP[5] study, the TW sequences similarly accounted for 50% of the calculated CDF. The most recent Peach Bottom studies (IDCOR and SARP) show a two-and-three-order-of-magnitude reduction, respectively, in TW sequences quantification. Therefore, a guideline has been developed in Section 4 to generically address the mechanisms already effectively employed at Peach Bottom to control the frequency of TW sequences (and other related loss of CHR removal sequences).

There are a number of factors associated with this reduction in the frequency of TW sequences. When the RSS study was performed, it was assumed that overheating of the suppression pool failed emergency core cooling (ECC) injection and therefore the containment failed with a conditional probability of unity. ECC injection failure came about either by failure to maintain NPSH conditions for the ECC pumps because of the heated pool conditions or, surviving that, loss of their suction source by some overpressure failure of the containment itself. Since that early study, investigations into ECC pump survivability have demonstrated that the pumps have a substantial likelihood of successful operation given a heated suppression pool. In addition, the containment failure concern is now mitigated by containment (wetwell) venting procedures at Peach Bottom (refer to Table 4.1). Alternate sources of

injection capability have also been established to preclude reliance upon an overheated suppression pool (refer to Table 4.6).

3.3.3 Reactor Pressure Vessel Depressurization Performance

As mentioned in Section 3.1.2, the ADS is an important system in mitigating loss of high-pressure injection sequences. Although neither SARP nor IDCOR indicate that these are dominant sequences for Peach Bottom, other studies (Limerick and Shoreham) indicate that failure to manually depressurize can be an important contributor to core melt (about 6×10^{-5}/reactor year). Guidelines and criteria are provided in Table 4.7 to ensure that other Mark I plants have a low TQUX frequency. Additionally, the ability to depressurize the RPV is an important mitigative feature that helps maintain suppression pool effectiveness (Guideline 2).

3.3.4 Support System Interdependencies

Most PRAs have stressed the importance of unrecognized interdependencies having the potential to compromise the performance of many critical safety systems. In many cases, risk assessment studies have identified such vulnerabilities very early in the study and "fixes" have been made which substantially reduced risk. Although no such dependency-caused vulnerability has been identified for Peach Bottom, "engineering judgement" indicates that it may be useful to search for the existence of such interdependencies in other Mark I plants.

3.3.5 Flooding Within the Reactor Building

One of the accident sequences, whose potential for contributing to the core damage frequency was specifically evaluated in the Shoreham Nuclear Power Station (SNPS) PRA,[6] is the release of excessive water into the reactor building. Both the SNPS PRA and the Brookhaven National Laboratory (BNL) review[7] of the SNPS PRA revealed that accident sequences induced by such an initiator contribute substantially to the CDF (3.9×10^{-6} and 2.0×10^{-5}/reactor year, respectively).

To help ensure that Mark I plants which may have a similar safety-related equipment flooding potential can be identified, a guideline and associated criteria are provided in Section 4.3 which may be used to screen for such vulnerabilities.

3.4 References for Section 3

1. "Peach Bottom Atomic Power Station-Integrated Containment Analyses," IDCOR Technical Report T23.1PB, March 1985.

2. C. N. Amos et al., "Containment Event Analysis for Postulated Severe Accidents: Peach Bottom Atomic Power Station, Unit 2," Sandia National Laboratories, NUREG/CR-4700, Vol. 3, Draft Report for Comment, May 1987.

3. G. A. Greene, K. R. Perkins and S. A. Hodge, "Impact of Core-Concrete Interactions in the Mark I Containment Drywell on Containment Integrity and Failure of the Drywell Liner," Paper IAEA-SM-281/36, Presented at the

International Symposium on Source Term Evaluation for Accident Conditions, Columbus, Ohio, November 1985.

4. "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, NUREG-75/014, October 1975.

5. S. E. Mays et al., "Interim Reliability Evaluation Program: Analysis of the Browns Ferry, Unit 1, Nuclear Plant," Idaho National Engineering Laboratory, NUREG/CR-2802, July 1982.

6. "Probabilistic Risk Assessment - Shoreham Nuclear Power Station," Long Island Lighting Company, June 1983.

7. D. Ilberg et al., "A Review of the Shoreham Nuclear Power Station Probabilistic Risk Assessment," Brookhaven National Laboratory, NUREG/CR-4050, June 1985.

4. GUIDELINES AND CRITERIA FOR A BWR WITH A MARK I CONTAINMENT

In Section 3, those accident sequences that dominate the core-damage frequency (CDF) were identified as were those that are potentially of high consequence. Vulnerabilities of the Mark I containment to severe accident containment loads were discussed and those features of a BWR with a Mark I containment, which are important for preventing core damage and are available for mitigating fission-product release to the environment were identified.

Based on the "insights" from previous PRA studies and other severe accident research, the following sections provide guidelines defining "deterministic, plant-specific guidance on the design features and operating characteristics which are to be examined by the utilities,"[1] and criteria defining "deterministic standards for judging the acceptability of plant features."[1] From SECY-86-76,[2] further guidance is provided in defining guidelines and criteria. These guidelines "will specify the plant features and operator actions which are considered important to ensuring acceptable risk for the reference plant."[2] Further acceptance criteria (for the various guidelines) "will specify the attributes necessary to ensure acceptable performance."[2]

Based on this work, nine guidelines were developed which reflect the importance of these features to plant risk. As discussed in Section 2.2.1 these guidelines indicate areas of potential improvements for various areas of plant design and operation of which utilities should be aware when conducting assessments. It is further noted that a number of the guidelines appear to overlap various generic issues as defined by the NRC. Final resolution and disposition of these generic issues may encompass NRC-imposed requirements. However, the guidelines and criteria presented herein are intended only for the purposes noted above. The guidelines are summarized in Table 1.1.

Guidelines 1 and 2 were developed to ensure the capability to mitigate fission-product releases (Goal 1) with reference to maintaining containment integrity and maintaining suppression pool effectiveness.

Guidelines 3 and 4 were developed for controlling the frequency of high-consequence sequences (Goal 2) with reference to minimizing interfacing systems LOCA frequency and mitigating anticipated transients without scram (ATWS) sequences.

Finally, Guidelines 5 through 9 were developed for reducing high core-damage frequency sequences (Goal 3) with reference to mitigating station blackout (SBO) sequences, mitigating loss of containment heat removal sequences, enhancing reactor pressure vessel (RPV) depressurization performance, examining support system interdependencies, and mitigating floods within the reactor building.

The remainder of this section is organized into three subsections corresponding to the three basic goals. In each subsection, the corresponding guidelines are discussed from which detailed criteria are developed in order to provide standards by which each plant could be measured for compliance with the guidelines. The criteria address (see Section 2.2.2), under severe accident conditions, the general issues of (1) survivability of equipment (i.e., whenever credit is given for a system or a component to mitigate the accident, the ability of the equipment to function under the environmental conditions

and fluid dynamic loads associated with severe accident sequences must be taken into account), (2) equipment capabilities, capacities, and duration of operability, (3) accessibility of equipment, (4) availability of support systems, (5) identification of necessary components, (6) identification of important operator actions, and (7) identification of parameters for initiation of mitigating systems and operator actions.

4.1 Mitigate Fission-Product Releases

For a BWR that has a Mark I containment, the dominant core-damage sequences were found to be station blackout (TB) and anticipated transients without scram (ATWS). In order to minimize offsite consequences for the sequences, the containment systems (both primary and secondary) should be able to retain a substantial fraction of fission products released even under these severe accident conditions.

4.1.1 Maintain Containment Integrity and Suppression Pool Effectiveness (Guidelines 1 and 2)

As discussed in Section 3, the most important systems for mitigating high-consequence sequences are the containment and its suppression pool. In addition to condensing the steam generated in an accident, the suppression pool also acts to remove fission products from the containment atmosphere. As long as any release path is forced through the pool (e.g., during wetwell venting), the pool will act to reduce the environmental release fractions by a factor of 10 or more. Thus, the miligative guideline deals with maintaining containment integrity, ensuring the effectiveness of the pool as a fission-product mitigation system.

Tables 4.1 and 4.2 provide criteria which may be used to evaluate each plant's capability to avoid breach of the containment and suppression pool bypass and possible suppression pool bypass mechanisms that were identified in Section 3.

4.2 Control the Frequency of High-Consequence Sequences

In Section 4.1, guidelines and criteria were developed that should effectively ensure containment integrity and mitigate fission-product releases for the broad classes of accident sequences that were found in Appendix A to be important to the core-damage frequency. However, two accident sequences were identified in Appendix A for which the BWR Mark I containment has limited means of mitigating fission-product releases; namely an interfacing systems LOCA and an ATWS with a power transient. In this section, guidelines and criteria for controlling the frequency of occurrence of these potentially high-consequence sequences are developed.

4.2.1 Interfacing Systems LOCA (Guideline 3)

In general, BWR Mark I PRAs have found the interfacing systems LOCA (ISL) to be a highly unlikely event (less than 10^{-7}/reactor year). However there are some BWRs (e.g., Shoreham) for which the ISL is risk significant because of the potentially high releases. The objective of this guideline and associated criteria is to ensure that the frequency of ISL events is kept at an acceptably low level. Brookhaven National Laboratory (BNL) is presently

performing a study to provide technical support to the NRC for the meaningful resolution of the generic issue related to ISL (GI-105). Therefore, the criteria for this guideline should be considered appropriate pending resolution of the generic issue.

In order to control the frequency of ISL sequences, specific performance criteria have been developed to assess the performance of equipment, systems and operators. The criteria relate to equipment (low-pressure systems interfacing with high-pressure systems) and operator performance (isolation and relief valve maintenance and surveillance).

Detailed criteria developed for this guideline are given in Table 4.3.

4.2.2 Anticipated Transients Without Scram (Guideline 4)

The important attributes of the ATWS sequence with respect to operator actions were found[3] to be the likelihood of misleading instrumentation, the need to inhibit automatic safety systems, the use of required mitigating actions which conflict with operator response to other accident conditions, and the need for coordinated actions and communications among control room crew members under highly stressful conditions.

For the ATWS guideline, the performance of equipment, systems, and operators should be assessed against specific performance criteria to ensure successful use of this guideline. The criteria relate to the equipment, systems, and operator performance by emphasizing operator familiarization, aids, and understanding of potentially conflicting signals.

Detailed criteria developed for this guideline are given in Table 4.4 and are based upon the assumption that each of the plants is (or will be) in compliance with the NRC rule on "Reduction of Risk from Anticipated Transients Without Scram for Light-Water-Cooled Nuclear Power Plants."[4]

4.3 Reduce High Core-Damage Frequency Sequences

The major contributors to the core-damage frequency (CDF) are presented in Appendix A. The IDCOR and ASEP/SARP analyses indicate that the SBO and ATWS sequences are the dominant contributors to the CDF. The results of other PRAs and PRA reviews indicate that in addition to those two types of sequences, other sequences, namely, loss of containment heat removal (CHR) sequences (TW, SI, and TPQI sequences) and sequences with failure to depressurize the reactor pressure vessel (RPV) for injection with low-pressure systems (TQUX sequences), can also be major contributors to the CDF.

4.3.1 Station Blackout (Guideline 5)

In most PRAs for light-water-reactors (LWRs), station blackout (SBO) sequences have been major or prominent contributors to the CDF. As part of the effort to resolve the unresolved safety issue (USI A-44), the NRC is proposing to amend its regulations "to provide further assurance that an SBO (loss of both offsite power and onsite emergency ac power systems) will not adversely affect the public health and safety."[5] For accident sequences, developed by an individual plant examination (IPE), which involve the loss of offsite power and onsite emergency power, the proposed SBO rule should be examined for

applicability. The criteria associated with Guideline 5 are intended to emphasize the need to search for plant specific features and potential common cause failures which could disable systems required to work during an SBO. For individual plants which are found to have a vulnerability to SBO, the criteria given in Table 4.5 highlight the importance of proper emergency procedures and operator training in recovering from an SBO event.

4.3.2 Loss of Containment Heat Removal (Guideline 6)

For some of the PRAs and the PRA reviews used in this study, sequences with successful coolant injection but with subsequent loss of containment heat removal (CHR) (TW, SI, and TPQI sequences in Tables A.1 and A.5) can be important contributors to the CDF; in those PRAs it is assumed that containment failure causes loss of ECC injection. As discussed in Section 3.3.3, in the PRAs where those sequences are not important, the main factor for the low contribution to CDF is because of credit given for containment venting and alternative sources of injection. Therefore it appears to be important to have alternative injection sources available in addition to wetwell venting to provide adequate CHR during accident sequences with successful ECC injection but with subsequent loss of CHR. Previous studies have shown that injection is not lost in TW sequences for 20 or more hours. Thus alternate injection sources should be sized to remove decay heat at that time (~0.5%).

Detailed criteria developed for this guideline are given in Table 4.6.

4.3.3 Reactor Pressure Vessel Depressurization Performance (Guideline 7)

In some of the PRAs examined in this study, sequences with failure to depressurize the reactor pressure vessel (RPV) after failure of the high-pressure injection systems (TQUX sequences) are important contributors to CDF. In all these PRAs, the automatic actuation of the ADS only occurs on coincident signals of "high" drywell pressure and "low" reactor vessel water level held for a time delay of 2 minutes. For a large number of transients with loss of high-pressure injection, these coincident signals will not occur. Therefore, the contribution of these sequences to CDF is dependent upon the intervention by the operator to manually depressurize the reactor. In Table A.6, it can be seen that these probabilities vary from 1.8×10^{-4}/demand to 6.0×10^{-3}/demand. In Peach Bottom, the CDF was reduced by changing the logic of the ADS auto-actuation logic to eliminate the need for the "high" drywell pressure and changing the water level time-initiation setpoint to "low-low" and the initiation time delay to 8 minutes.

Detailed criteria developed for this guideline are given in Table 4.7.

4.3.4 Support System Interdependencies (Guideline 8)

One of the primary benefits of performing a rigorous PRA is that the system interdependencies are modeled and are reflected in the results. However, not all PRA studies have performed rigorous interdependence analyses and, therefore have not ferreted out all of the possible subtle interdependencies. This may have profound effects upon their results. A dependency is defined as the failure of one system leading directly or indirectly to the failure of another system.

An in-depth application of basic PRA methodology with respect to interdependencies yielded significant findings on a previously heavily studied PWR. To illustrate this point, reference is made to the BNL study of system interactions (support system interdependencies) at Indian Point Unit 3.[6] The major finding of that study was that a specific single station emergency battery could fail and, among other things, negate the entire low-pressure injection function. The point to be emphasized here is that none of the numerous other studies and reviews of the Indian Point 3 design were able to detect this important single failure nor did the BNL study until all the support systems were explicitly modeled, linked together (the fault tree linking approach[7]) and solved using the SETS computer code.[8]

NUREG-1150[9] has provided a thorough application of the latest PRA methods to five reference plants and the results point out numerous insights into the importance of specific design differences among the studied plants. However, the NUREG-1150 authors emphasize the importance of support system differences and the difficulty of extrapolating the result from one plant to another.

It is not sufficient to make a single overall dependency table of the front-line and support systems for a given plant and simply compare that to the reference plant. No two plants will have the same set of system interdependencies. Support systems vary widely from plant to plant even though the plants may be of a similar class and have the same set of front-line systems.

It is recognized that following the steps outlined in Table 4.8, in a rigorous fashion, is a major undertaking. This fact, however, does not diminish its importance.

Based upon the dominance of the SBO sequences for Peach Bottom as well as for other BWR designs, it also is recommended that a detailed interdependency table be constructed for this sequence with all dependencies conditioned upon the existence of an SBO for various lengths of time. This table should also explicitly identify all of the expected failure mechanisms (e.g., identify whether battery failure is because of loss of room cooling or charge depletion).

4.3.5 Flooding Within the Reactor Building (Guideline 9)

Although medium or small leakages can be adequately mitigated by the existing sumps or pumpback systems, large water leakages are of primary concern in reactor building (RB) flooding. Potential water sources for excessive water release into the lowest RB level include the suppression pool, the condensate storage tank, the reactor coolant system, the service water system and the fire protection system storage tank. Some of the major equipment located in the lowest RB level compartment may include emergency core cooling (ECC) pumps and their electrical control panels for the high-pressure coolant injection (HPCI), RCIC, core spray and low-pressure coolant injection (LPCI) systems.

RB flooding can be initiated by (1) a major maintenance which requires exposing a safety system to the RB atmosphere, and (2) breaks in the pressurized or the non-pressurized part of piping or components. In item 1, "major maintenance" refers to those actions which would require dismantling of system components thus eliminating a barrier between large sources of water and

the RB. RB flooding can partly be prevented and/or mitigated through proper training and procedures. For example, once the RB is flooded, the operator should be able to follow the instructions for responding to the alarm to identify the source of the flood and isolate it before the water level in the lowest compartment reaches a critical level. The operator should also know about alternative devices or equipment which can be utilized to provide coolant injection to the RPV in case of emergency core cooling (ECCS) systems equipment failures in the flooded compartment.

The BNL study[10] of the Shoreham Nuclear Power Station (SNPS) revealed that although the SNPS Alarm Response Procedures give general guidelines for monitoring system parameters to determine the leakage location and initiate the leakage isolation, specific requirements for operators to systematically check the operation parameters of relevant systems are not included. BNL also identified that the random failure of an equipment protection electric circuit breaker coinciding with RB flooding may result in the propagation of failures to the upstream motor control center (MCC), other MCCs, and the associated load centers. It is important that this or similar potential common-mode failures be avoided.

Although this type of vulnerability to flooding was identified for a Mark II plant, it is believed that the concerns are of general applicability to other designs. Thus, it is recommended that Mark I plants also be screened for flooding vulnerabilities.

Detailed criteria developed for this guideline are given in Table 4.9.

4.4 Using the Guidelines and Criteria

Numerous investigations, including PRAs, have been performed for the reference plants and for similar plants by both the NRC and the nuclear power industry. The insights gained from many of the studies have been used in developing the guidelines and criteria contained in this report (including the other volumes relating to other plant types). The guidelines and criteria are issued to guide the analyst performing an IPE. This guidance is in the form of plant features, operator actions and the criteria for assessing those features and actions found to be helpful in reducing the overall risk for Peach Bottom and other Mark I plants. Thus, the guidance is given to provide a resource in examining the subject plant to determine if the same, or similar, guidelines will be of value in reducing overall plant risk. These guidelines and criteria are intended to be used solely as guidance, but they may include (as a subset) some requirements generated by the NRC on generic issues.

4.5 References for Section 4

1. R. Barrett, "Status of the Severe Accident Program for Operating Reactors," U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation Staff Presentation to the ACRS Subcommittee on Class 9 Accidents, February 24, 1986.

2. SECY-86-76, "Implementation Plan for the Severe Accident Policy Statement and the Regulatory Use of New Source-Term Information," NRC/EDO, February 28, 1986.

2. SECY-86-76, "Implementation Plan for the Severe Accident Policy Statement and the Regulatory Use of New Source-Term Information," NRC/EDO, February 28, 1986.

3. W. Luckas et al., "A Human Reliability Analysis for the ATWS Accident Sequence With MSIV Closure at the Peach Bottom Atomic Power Station," Brookhaven National Laboratory, Technical Report A-3272, April 1986.

4. ATWS Final Rule - Code of Federal Regulations, Title 10, Section 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram Events for Light-Water Cooled Nuclear Power Plants," June 1984.

5. NRC Station Blackout Proposed Rule, Federal Register, Volume 51, No. 55/March 21, 1986, pgs. 9829-9835.

6. R. Youngblood et al., "Fault Tree Application to the Study of Systems Interactions at Indian Point 3," NUREG/CR-4207, January 1986.

7. American Nuclear Society and Institute of Electrical and Electronics Engineers, "A PRA Procedures Guide," NUREG/CR-2300, January 1983.

8. R. B. Worrell and D. W. Stack, "A SETS User's Manual for the Fault Tree Analyst," Sandia National Laboratories, NUREG/CR-0465, SAND77-2051, November 1978.

9. "Reactor Risk Reference Document," U.S. Nuclear Regulatory Commission, NUREG-1150, Draft for Comment, February 1987.

10. K. Shiu et al., "A Review of the Accident Sequences Following an Excessive Release of Water at Elevation 8' of the Reactor Building in the Shoreham Nuclear Power Station," Draft Report, NUREG/CR-4049, BNL-NUREG-51835, March 1985.

Table 4.1 Criteria for BWR Mark I Containment Guideline 1: Maintain Containment Integrity

Concern: Breach of the containment boundary in the progression of a severe accident can lead to significant releases of radioactivity.

Functions: Wetwell Venting of Noncondensable Gases - (Guideline 1.A)

Guideline 1.A. Provide Wetwell Venting

Basis: Implementation of wetwell venting will significantly reduce the potential for loss of containment integrity because of overpressurization events.

Caution: Containment venting should not be indiscriminately performed. A clear understanding of the accident sequence in progress should have been attained before initiating venting. The effects of venting should have been assessed and made known to the operators during the training program. The assessment should include the effects of containment venting on the operation of ECC injection systems and health consequences.

Criteria:

1.A.1. For accident sequences where wetwell venting has been assessed to be beneficial, wetwell venting should commence, except for a station blackout, when containment pressure reaches the predetermined containment venting pressure setpoint. In selecting the containment venting pressure setpoint, the following functions should be ensured:

a. The ultimate containment pressure capability would not be exceeded.

b. The backpressure acting on the safety-relief valve assemblies would not prevent them from performing their function.

c. The vent valve assemblies would not be prevented from performing their function.

During a station blackout, wetwell venting should commence in accordance with the criteria developed using the BWR Emergency Procedure Guidelines (EPG), i.e., following the onset of the transient and before depletion of the station batteries. If station batteries are not available, manual initiation of wetwell venting should be assessed (see Criterion 1.A.2).

1.A.2. If manual initiation of wetwell venting is deemed necessary, the time required to perform this function should be taken into account in the training and procedures to preclude the potential for exposing personnel to the harsh environment. Otherwise, the containment venting valve(s) should be capable of being remotely actuated during a station blackout.

Table 4.1 (Continued)

1.A.3. Operator training and emergency procedures should specify the plant parameters that will prompt the operators to make preparation, commence and terminate the venting sequence. The training and procedures should also be consistent with the required actions and timing of those actions so venting will commence immediately when required (see Criteria 1.A.1 and 1.A.2). The training and procedures should further specify the flowpath(s) available for venting, specific components to be aligned, and the required positions/states for these components. The training and procedures should specify how to proceed if it is not possible to terminate venting.

1.A.4. For each accident sequence where venting is credited (e.g., assumed to prevent containment failure) the capacity of the vent lines and associated vent valves should be assessed to determine whether the venting capacity has the capability to decrease containment pressure.

1.A.5. The criteria for filtering are dependent on the potential for bypassing the suppression pool. Whether the suppression pool is bypassed or not, the radiological release should be reduced by an order of magnitude compared to no filtering. The venting flowpath should ensure that all media to be vented pass through the suppression pool thus providing filtering by the pool. If the potential of a vent path to bypass the suppression pool is high, a filter should be provided in this vent path with the ability to reduce radiological releases by an order of magnitude.

1.A.6. Equipment designated to support wetwell venting should be assessed for its ability to function reliably for a sufficient period under the predicted environmental and fluid loads associated with venting commencement pressure. If necessary, it should be enhanced to include operation during the vaporization release phase of core-concrete interaction.

1.A.7. The effects of possible hydrogen burn, radiation, and/or steam on equipment located in the reactor building outside of the primary containment should be considered in the venting assessment. If equipment important to the mitigation of accident sequences is jeopardized by venting, alternate venting paths, judged not to be detrimental, should be identified and assessed. Consideration should also be given to the effectiveness of the reactor building blowout panels and fire sprays to accommodate the discharge through the primary containment vents, thereby ensuring reactor building structural integrity.

1.A.8. The effects of possible containment depressurization on the NPSH of the ECC related pumps should be assessed. Alternate injection sources which are unaffected by venting should be considered.

1.A.9. The capability to terminate venting and the conditions under which venting would be terminated should be considered in the venting assessment. Specifically, the level of radioactivity in the wetwell airspace should be considered with regard to the projected offsite consequences.

Table 4.1 (Continued)

1.A.10.	Operator training and emergency procedures should specify the possible actions to preclude deinerting the containment by terminating venting before a negative pressure differential is reached in the vent path.

Table 4.2 Criteria for BWR Mark I Containment Guideline 2: Maintain Suppression Pool Effectiveness

Concern: Bypass of the suppression pool in the progression of a severe accident can lead to significant releases of radioactivity that would otherwise not occur if the fission products were retained in the containment and/or scrubbed by the suppression pool.

Functions: Suppression Pool Bypass (Guideline 2.A)
- Debris Confinement

Drywell Spray (Guideline 2.B)
- Containment Heat Removal
- Fission Product Scrubbing
- Debris Bed Cooling

Guideline 2.A. Prevent Suppression Pool Bypass

Basis: Implementation of the following criteria will significantly reduce the potential for bypassing the suppression pool.

Criteria:

2.A.1. In view of the uncertainty about the amount and timing of core debris leaving the reactor pressure vessel (RPV), an assessment should be made of the fraction of the core that can be controlled or confined to prevent contact with the steel containment shell.

Guidance: If 100% of the core cannot be controlled or confined, additional debris control methods should be considered such as a concrete or magnesium oxide curbing of sufficient height to prevent or delay the debris from reaching the containment shell and confine the debris in an area that can be cooled. The curbing should be anchored in place to resist the hydrostatic head of the core debris and should have sufficient strength and thickness to withstand core-debris attack.

2.A.2. Appropriate maintenance, surveillance, and emergency operating procedures and training should specify the actions to be taken (and intervals at which these actions are to be performed) to ensure that the containment isolation valves and vacuum breakers are capable of closing as required and remaining closed during severe-accident conditions.

2.A.3. The projected leakage rate through the main steam isolation valves (MSIVs) should be assessed for its source term contribution in the dominant accident sequences.

2.A.4. The effect of reactor coolant system leakage accumulation and associated radioactivity in the drywell following isolation of equipment drainlines should be assessed for their source term contribution to sequences in which drywell venting or failure is anticipated.

Table 4.2 (Continued)

Guideline 2.B. Provide Drywell Spray

Basis: Implementation of the following criteria should aid in decontaminating the drywell atmosphere of fission products, should help control the containment pressure rise because of the decay heat load, and should promote debris cooling.

Criteria:

The following should be assessed to ensure containment heat removal capability:

2.B.1. The heat removal provided by the drywell-spray-related components should be sufficient to remove heat loads anticipated during the dominant accident sequences. These loads include but are not limited to decay heat and the chemical energy released from metallic oxidation.

2.B.2. Drywell spray should commence when the containment pressure exceeds a predetermined value calculated in accordance with the BWR EPG or before the drywell temperature reaches the value at which ADS is qualified.

2.B.3. Drywell spray should be terminated when the containment pressure decreases below a predetermined value calculated in accordance with the EPG.

Guidance: Alternate sources of drywell spray such as a diesel-driven fire pump should be assessed for their capability to provide sufficient flow and head for adequate containment heat removal.

2.B.4. The ability of equipment designated for drywell spray to function in a reliable manner under the predicted containment conditions associated with sequences for which operation of the containment spray is needed should be assessed.

2.B.5. Operator training and emergency procedures should specify the flowpaths and specific components to be aligned and their required positions for initiating drywell spray. If a backup system and/or equipment is to be utilized, operator training and procedures should identify the flowpaths and specific actions required for any temporary system cross-connections.

2.B.6. Operator training and emergency procedures should specify the plant parameters that will prompt the operators to initiate and terminate the drywell spray. Training and procedures should be consistent with the time required to align the system and components as required.

The following should be assessed in order to evaluate the capability of the drywell spray to reduce fission-product contamination of the drywell atmosphere:

Table 4.2 (Continued)

2.B.7.	The spray system should be assessed for its ability to cover the entire drywell volume for an adequate time, with spray droplets of an appropriate size. Such an assessment would include the total amount of water available for long-term spray operation, as well as the pressure under which the water could be supplied to the spray headers by various sources. The elevation of the spray in the drywell, the nozzle spray pattern, as well as large obstructions in the drywell below the spray headers, should be considered when assessing volume coverage.
In addition, to promote debris bed cooling the following criteria should be assessed to ensure flooding of the drywell via sprays before RPV failure:	
2.B.8.	The spray initiation point (Criterion 2.B.2) should be assessed to ensure that the sprays will be initiated early enough to flood the drywell floor before RPV failure for the dominant accident sequences.
2.B.9.	The spray termination point (Criterion 2.B.3) should be assessed to ensure that spray termination will not allow the debris bed to reheat after it has been quenched.

Table 4.3 Criteria for BWR Mark I Containment Guideline 3: Interfacing Systems LOCA

Concern: Although the interfacing systems LOCA sequences are not considered to be leading contributors to core-damage frequency, they represent potentially high release sequences and they appear to contribute significantly to the overall risk for the plant under review.

Function: Maintain Reactor Coolant System Integrity

Guideline 3.A. Prevent Overpressurization of Low-Pressure Systems

Basis: Implementation of the following criteria will ensure that the frequency of an interfacing systems LOCA will remain acceptably low.

Criteria:

Note: Resolution of Generic Issue 105 (GI-105), which deals with interfacing systems LOCAs for both BWRs and PWRs, may have an impact on this guideline. Therefore, the criteria below should be considered as appropriate pending resolution of GI-105.

3.A.1. All low-pressure lines that potentially could be overpressurized should be identified and should be provided with alarms to alert the operator to the symptoms of an overpressure event.

3.A.2. The equipment designated to provide isolation and prevent overpressurization, such as the RHR line isolation valves or the low-pressure injection system check valves, should periodically undergo operability testing and local leak rate testing (LLRT).

3.A.3. The relief capability of the relief valves designated to mitigate low-pressure system overpressurization should be established. In most if not all cases, these relief valves were not sized with the possibility of an interfacing systems LOCA in mind. However, given that an interfacing systems LOCA occurs in a non-isolatable portion of a low-pressure system, there may be alternatives available to the operator such as taking advantage of additional relief valves. If such or similar actions are found to be helpful, they should be factored into the appropriate emergency procedures.

3.A.4. Operator training and procedures should specify the actions to be taken to isolate the low-pressure systems identified above or to depressurize the primary system, thereby mitigating the consequences of the interfacing systems LOCA.

3.A.5. After each reactor shutdown and cooldown, the isolation function of the pressure isolation valves should be tested. These valves should not be tested under reactor operating conditions.

Table 4.4 Criteria for BWR Mark I Containment Guideline 4: Anticipated Transients Without Scram (ATWS)

Concern: ATWS sequences have been shown to be one of the leading classes of severe-accident sequences both in terms of core-damage frequency and risk.

Function: Operator Response During ATWS (Guideline 4.A)

Guideline 4.A. Provide Operator Response During ATWS

Basis: The criteria developed here are based on the assumption that each of the plants is (or will be) in compliance with the ATWS final rule dated July 26, 1984. PRA studies have shown that the predicted core-damage frequency because of ATWS is significantly lowered based upon modifications which comply with the ATWS rule. The major thrust of the ATWS rule is on the addition and/or upgrading of scram related systems and equipment to prevent an ATWS. Human reliability studies performed in support of NUREG-1150 point to potential benefits for improved operator training and procedures to mitigate the effects of an ATWS and prevent core damage from occurring. For any individual plant which may be found to be vulnerable to ATWS, the following criteria reflect added measures that emphasize the operator's role and function in mitigating an ATWS initiator.

During an ATWS sequence, the operator is required to inject boron into the reactor pressure vessel (RPV) to inhibit initiation of automatic safety systems and to attempt to manually control and mitigate the outcome of the event. In contrast, most other accident sequences are prevented or mitigated by systems which allow the operator to monitor automatic system initiation and require intervention only when a system fails to function adequately. Thus, an ATWS sequence requires operator responses that are in opposition to the highly trained responses required for the recovery and mitigation of all other off-normal and accident events. Therefore, operator training and procedures for the ATWS sequences should specifically prepare operators to perform the unique ATWS actions called for in the BWR EPG as well as in the other measures below.

Criteria:

4.A.1. Operator training and emergency procedures should specify the plant parameters that are indicative of ATWS and the actions to be taken to verify that the reactor coolant recirculating pumps have tripped automatically. Additionally, they should specify the actions to be taken if the reactor coolant recirculating pumps do not trip automatically.

4.A.2. Operator training and emergency procedures should ensure that standby liquid control system (SLCS) injection is initiated manually, as required, during an ATWS. Operator training and procedures should also specify the plant parameters that indicate manual SLCS actuation and the actions to be taken and verification to be made to ensure that the SLCS was actuated.

4.A.3. Operator training and procedures should ensure operator familiarity with reactor water level control during ATWS.

Note: This unique control requires actions that conflict with mitigating actions for all other accidents that call for flooding the RPV to ensure the reactor core is covered.

4.A.4. Since RPV water level indicators may be inaccurate and may provide conflicting indications of the water level, operator training and procedures should provide guidance to the operator.

4.A.5. The automatic depressurization system (ADS) should be capable of being defeated by the operator during an ATWS before its automatic initiation. Operator training and procedures should address the possible reluctance of operators to defeat a safety system, in particular, the need to inhibit the ADS immediately after an SLCS initiation attempt.

4.A.6. Operator training and procedures should specify the responsibilities of operating staff crew members and clarify how information will be exchanged among them. In particular, instrumentation readings may have to be relayed between the crew member(s) operating the control boards and the senior reactor operator coordinating the crew's response to the accident.

4.A.7. The capability, of the systems and equipment required for mitigating an ATWS, to function under predicted environmental and fluid loads associated with severe-accident sequences should be assessed to determine whether the equipment would be available for an appropriate time of operation.

4.A.8. An assessment should be made of the feasibility of establishing the main condenser as a heat sink by reopening the main steam isolation valves and turbine bypass valves, if possible.

Table 4.5 Criteria for BWR Mark I Containment
Guideline 5: Station Blackout

Concern: Station blackout sequences have been shown to be one of the leading classes of severe-accident sequences both in terms of core-damage frequency and risk.

Functions: Reactor Pressure Vessel Injection (Guideline 5.A)
Containment Integrity (Guideline 1)

Guideline 5.A. Provide Reactor Pressure Vessel Injection

Basis: Significant study and research have preceded current work on severe accidents; in particular, reference is made to the rulemaking activity already under way on station blackout. It is assumed that when the station blackout rule is finalized, some requirements of the rule may be similar in form to the criteria below. Nevertheless, during an individual plant examination (IPE), it is important to highlight those areas which previous PRAs found to be important contributors to the station blackout core-damage frequency. For those specific plants which are found to be vulnerable to station blackout events, the criteria below will assist in identifying potential areas for plant improvements as well as identifying operator actions which are key to mitigating a station blackout event.

Criteria:

5.A.1. For the BWR-4 design, the high-pressure coolant injection (HPCI) system and the reactor core isolation cooling (RCIC) system, are two systems intended for the purpose of RPV injection independent of ac power. However, it has been postulated that these systems cannot continue operating in the presence of a prolonged blackout. Therefore, the HPCI and RCIC should be assessed with respect to extending their capability to function in the presence of a station blackout.

5.A.2. Operator training and procedures should specify the plant parameters indicative of HPCI and RCIC initiation. Additionally, the training and procedures should specify the actions required to place in operation and/or ensure continued operation of these systems under station blackout conditions.

5.A.3. The capability of HPCI and RCIC systems to function under predicted environmental conditions and fluid loads associated with station blackout should be assessed to determine whether they are available for an appropriate time of operation.

5.A.4. Special emphasis should be placed upon the review of the dc power systems and other emergency power support systems to ensure that common cause failures have been eliminated to the extent practical from the design.

Table 4.6 Criteria for BWR Mark I Containment
Guideline 6: Loss of Containment
Heat Removal

Concern: Failure to remove the decay heat buildup in the suppression pool (loss of containment heat removal) following a transient event has been shown to create NPSH problems for the pumps taking suction from the suppression pool and, therefore, can lead to injection failure, subsequent core damage, and containment failure. The RSS indicated that this was a leading class of core-damage sequences.

Functions: Long-Term Emergency Core Cooling (ECC) (Guideline 6.A)
Maintain Containment Integrity (see Guideline 1)

Guideline 6.A. Provide Long-Term Emergency Core Cooling

Basis: Implementation of the following criteria will ensure that the failure potential of ECC injection because of loss of the containment heat removal function (i.e., TW sequences). Maintenance of containment integrity is addressed by Guideline 1.

Criteria:

6.A.1. Operator training and procedures should specify methods and actions for heat removal via alternate injection path(s), in conjunction with wetwell venting for severe-accident conditions when suppression pool temperature precludes use of primary (ECC) injection paths.

6.A.2. For the alternate injection path(s), it should be demonstrated that the flow would be sufficient to preclude core damage.

6.A.3. If local operation of any equipment is required, the time required to perform these functions should be consistent with the time available to help prevent core damage and account for personnel exposure to the predicted severe-accident environment.

6.A.4. The capability of equipment used for alternate injection to function under the predicted environmental and fluid loads associated with severe-accident sequences should be assessed to determine whether the equipment would be available for an appropriate time of operation.

Table 4.7 Criteria for BWR Mark I Containment Guideline 7: Reactor Pressure Vessel (RPV) Depressurization Performance

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Concern: Sequences in which failure to depressurize the reactor pressure vessel (RPV) after failure of the high-pressure injection systems have been shown to be a leading contributor to the core-damage frequency. Many of these sequences do not create the condition necessary to actuate the ADS.

Function: RPV Depressurization (Guideline 7.A)

Guideline 7.A. Provide RPV Depressurization

Basis: Implementation of the following criteria will significantly improve the response of the ADS to facilitate depressurization of the RPV in order that low-pressure systems may be used to provide ECC injection.

Criteria:

7.A.1. The RPV should be capable of being automatically depressurized as may be required for all dominant accident sequences.

Guidance: An example of an acceptable modification to BWR-4 designs is the design change at Peach Bottom wherein the high drywell pressure signal was removed from the coincidence logic for automatic ADS actuation and the time delay for actuation has been lengthened. ASEP results for Peach Bottom indicated a high likelihood of concurrent ac and dc power failure. Plants with similar vulnerabilities may consider a dedicated backup dc supply to ensure depressurization capability under station blackout conditions.

7.A.2. The ADS should be capable of initiation and operation under the environmental conditions associated with the dominant accident sequences. In particular, the ADS should be capable of operating under the maximum pressure anticipated before venting (see Criterion 1.A.1).

7.A.3. Operator training and procedures must ensure reliable manual control of RPV depressurization.

7.A.4. The capability of the systems required to provide RPV depressurization (e.g., batteries and air supplies) to function under the predicted environmental conditions and fluid loads associated with severe accident sequences should be assessed to determine whether the equipment would be available for an appropriate time of operation.

Table 4.8 Criteria for BWR Mark I Containment
Guideline 8: Support System
Interdependencies

Concern: When conducting a PRA, IPE or similar analysis, it is imperative that the support system interdependencies be fully developed, understood, and reflected in the final results. Otherwise there is no assurance that the dominant core-damage/risk sequences have been identified.

Function: Support System Interdependencies (Guideline 8.A)

<u>Guideline 8.A. Examine Support System Interdependencies</u>

Basis: Implementation of the following criteria will ensure that the full set of support system interdependencies have been identified and have been reflected in the results.

Note: The following criteria are easily outlined but are not easily implemented. The complex nature of a nuclear power plant makes it imperative that this area of analysis be fully examined. However, since no two plants have identical support systems this analysis can only be done on a plant-specific basis.

Criteria:

8.A.1. All systems that provide any direct support to either a frontline or support system should be identified along with its supported system.

8.A.2. Each dependency should be conditioned as appropriate as to what sequences or under what (if not all) circumstances it applies. In view of its importance, a separate station blackout dependency table should be provided which gives the available systems, their anticipated survival period, and the ultimate cause (e.g., no room cooling) of their failure.

8.A.3. The dependencies should then be linked together (preferably by computer) within the analysis in order that the extent to which their influence reaches through the systems to a consequence will be discovered. This will help identify secondary dependencies to ensure that no one failure in a support system has any unknown critical outcome on other support or front line systems.

Table 4.9 Criteria for BWR Mark I Containment Guideline 9: Flooding Within the Reactor Building

Concern:	An excessive water release into a portion of the reactor building (RB) outside the primary containment which houses a concentration of safety-related equipment raises the possibility of a common-mode event disabling all the equipment in the compartment. At least one plant with a Mark II containment has been identified in which the location of safety equipment, including all ECCS pumps, in the lowest RB level makes the flooding initiator a substantial contributor to CDF.
Function:	Prevent or Mitigate RB Flooding (Guideline 9.A)

Guideline 9.A. Prevent or Mitigate Reactor Building Flooding

Basis: Implementation of the following criteria may reduce the potential of a common-mode failure of safety equipment because of RB flooding in Mark I containments where RB layout combines important safety equipment in low-lying portions of the RB with exposure to possible inundation.

Criteria:

9.A.1. Operator training and procedures should ensure that the operator will diagnose and isolate any flooding of the RB that occurs.

9.A.2. Operator training and procedures should ensure that the operator is prepared to use alternate injection sources still available if flooding causes a common-mode failure of ECCS equipment.

9.A.3. The electrical systems should be assessed for the possibility of cascading failures because of flood-induced electrical shorts. Additional isolation devices (e.g., circuit breakers) should be considered, if needed.

9.A.4. Water-tight doors in the lower levels of the RB should be alarmed to notify the operator when they have been left open.

APPENDIX A

SEVERE-ACCIDENT RISK INSIGHTS

In this appendix the IDCOR and SARP analyses for a BWR-4 with a Mark I containment are reviewed. Differences between the studies are identified and insights are identified which helped in the development of the plant-type specific guidelines for the reduction in the frequency of occurrence and mitigation of severe accidents which are discussed in Section 4 of this report. In addition to the IDCOR and SARP analyses, other studies relevant to a BWR-4 with a Mark I containment were also evaluated. The insights gained from these studies also contributed to the development of the guidelines and detailed criteria in Section 4

Section A.1 describes the review of the various estimates of the core-damage profile. Core-meltdown phenomena and containment response are addressed in Section A.2. Differences between IDCOR and SARP estimates of fission-product release and offsite consequences are discussed in Section A.3. Section A.4 addresses the potential offsite consequences of the severe accidents, which were found to be important. Finally, Section A.5 indicates the insights gained from the review of these studies.

A.1 Core-Damage Profile

The main objective of this section is to present, within the scope of this study, the Peach Bottom core-damage profiles emerging from the RSS,[1] IDCOR,[2] and ASEP/SARP[3] analyses. Also in this section, an attempt is made to clarify not only the most important differences in the three Peach Bottom studies, but also the differences between the Peach Bottom studies and the other PRAs considered in this study (Limerick PRA Review,[4] Browns Ferry IREP PRA,[5] and the Shoreham PRA Review[6]).

A.1.1 Peach Bottom Core-Damage Profiles - RSS, IDCOR, and ASEP/SARP

The first PRA (namely the RSS[1]) performed for Peach Bottom did not assemble its results for the core-damage frequency (CDF), but allowed the analyst to do so by using results presented in the RSS appendices. In the RSS, sequences initiated by a transient event and followed by failure to scram the reactor (ATWS sequences - 41% of the CDF) or by failure to provide long-term decay heat removal (DHR) from the containment (TW sequences - 53% of CDF) were found to be the most important contributors to the core-damage frequency. A total CDF of 3.2×10^{-5}/yr was obtained in the RSS and the breakdown by sequence-type is provided in Table A.1. Note that the RSS used median values for all system failures and mean values for the transient initiators.

The IDCOR study[2] presented two core-damage profiles for Peach Bottom, namely, the IDCOR baseline and the IDCOR committed core-damage profiles. In this analysis, only the IDCOR committed core-damage profile will be discussed.

The IDCOR committed core-damage profile was obtained by updating the RSS analysis. This updating process incorporated recent values for initiator frequencies, more realistic success criteria, plant modifications, and changes in the emergency procedures; all these changes are shown in Table A.2. It is important to note that for ATWS, TW and station blackout sequences new event

trees were developed in the IDCOR study; these new event trees were necessary to better characterize the progression of the sequences and the operator actions during the course of the accident.

The IDCOR committed core-damage profile is also presented in Table A.1. From this table, it can be seen that the total CDF is equal to 8.1×10^{-6}/yr and the most important contributors were determined to be transients with failure to scram (TC sequences - 90%) and station blackout sequences (TB sequences - 6%).

The results of the ASEP/SARP analysis for Peach Bottom (Table A.1) found that station blackout contributes 86% to the calculated core-damage frequency. The second leading contributor was ATWS with a 12% contribution. All other sequences combine to contribute to the remaining 2%.

In summary, the following can be stated about the relative contributions of the calculated accident sequences within each study:

(1) ATWS (TC sequences) were found to be important contributors in all three studies; they were the major contributors to the core damage in the IDCOR study.

(2) Station blackout sequences were the major contributors in the ASEP/SARP study; but were less important in the IDCOR study and negligible in the RSS.

(3) The TW sequences were only dominant in the RSS. The main reason for this difference comes from the inclusion of alternate methods for containment heat removal (containment venting with alternative injection sources such as the control rod drive pumps) in the IDCOR and ASEP/SARP analysis. The extended time frame (about 30 hours before core melt) and associated low decay heat level (about 0.5% of full power) imply a high likelihood of successful containment heat removal for these sequences.

The major differences in the dominant accident sequences are discussed in the next subsection.

A.1.2 Peach Bottom Dominant Sequences: Differences in RSS, IDCOR, and SARP Analysis

This section discusses the main differences present in the RSS, IDCOR, and SARP analyses for the following sequences:

- TC
- TB
- TW
- TQUV and TQUX

A.1.2.1 TC (ATWS) Sequences

Before discussing the differences in the CDFs for the various studies, the similarities and differences in the basic assumptions used in the studies have to be clearly identified:

(1) Scram failure was defined as failure of <u>all</u> of the 185 control rods to insert in all studies.

(2) A manually actuated standby liquid control system (SLCS) with a capacity of 43 gpm was assumed in the RSS. The IDCOR analysis and the ASEP/SARP analysis assumed an 86-gpm manually actuated SLCS, an alternate rod insertion (ARI), and RPV Level 1 MSIV closure.

(3) The current BWR emergency procedure guidelines were used only in IDCOR and SARP analyses.

The most important contributors to the core-damage frequency from TC sequences are discussed below for all of the three studies considered (RSS, IDCOR and SARP).

<u>RSS: TC = 1.3×10^{-5}/yr</u>

In the RSS, a very simple analysis was performed, and only one sequence was found to contribute, namely, any transient event (T = 10/y.) with failure of the reactor protection system (RPS) (1.3×10^{-5}) and failure of the operators to actuate the SLCS or to manually insert the control rods (probability = 0.1).

<u>IDCOR: TC = 7.3×10^{-6}/yr</u>

The IDCOR study presented a more detailed analysis of the TC sequences. In this analysis, the current BWR emergency procedure guidelines (e.g., maintain level at top of active fuel (TAF), control RPV injection with high- or low-pressure systems, avoid depressurization), as well as more recent results from ATWS deterministic analysis were used. Also, in the IDCOR analysis, if injection with low-pressure systems was successful, credit was given for containment heat removal by venting the containment. Even when the containment was assumed not to successfully vent (i.e., even if the containment fails), low-pressure injection was assumed to continue with a probability equal to 0.5.

In the IDCOR analysis, the transients were basically divided into two groups; namely, transients with the power conversion system (PCS) available (turbine trips - T_T) and transients with the PCS unavailable (isolation events - T_I). In addition, a distinction was made between transients at low power (<25%) and at high power (>25%).

Isolation transients (T_I; which include MSIV closure, inadvertently open relief valve (IORV), loss of FW, loss of condenser vacuum, and transfers from turbine trip events with early closure of MSIV) contributed about 58% to the ATWS CDF, and turbine trips (T_T) contributed the other 42%. The most important sequences were identified as:

$T_I^{>25\%}(1.03)*C_1(1.0\times10^{-5})*C_2(0.515)*VC_3(0.5)$ $\quad 2.65\times10^{-6}$ (36%)

$T_T^{>25\%}(4.19)*C_1(1.0\times10^{-5})*C_2(0.115)*VC_3(0.5)$ $\quad 2.41\times10^{-6}$ (33%)

$T_I^{>25\%}(1.03)*C_1(1.0x10^{-5})*\overline{C_2}(0.485)*U(0.23)*VC_3(0.5)$ $5.74x10^{-7}$ (8%)

$T_I^{<25\%}(0.222)*C_1(1.0x10^{-5})*C_2(0.215)*VC_3(0.5)$ $2.39x10^{-7}$ (3%)

$T_I^{>25\%}(1.03)*C_1(1.0x10^{-5})*C_2(0.515)*\overline{VC}_3(0.5)*\overline{VENT}(0.9)*E(0.1)$ $2.39x10^{-7}$ (3%)

A sensitivity analysis was performed by BNL to calculate the ATWS CDF with no credit given for containment venting and assuming that if containment fails, core damage would result. With those two assumptions the calculated CDF increased from $7.3x10^{-6}$/yr to about $1.2x10^{-5}$/yr.

ASEP/SARP: TC = $1x10^{-6}$/yr

The Peach Bottom ASEP/SARP study presented a more detailed quantification of the ATWS sequences (than either the RSS or the IDCOR studies). The ASEP/SARP study analyzed all the operator actions required during the progression of ATWS sequences. The Peach Bottom Transient Response Implementation Plan procedures were used in the ASEP/SARP quantification of the failure probabilities of the operator actions. Five operator actions, namely, SLCS actuation, inhibit ADS, high-pressure RPV level control, controlled depressurization, and low-pressure RPV level control, were analyzed in detail.[7]

The ASEP/SARP analysis obtained an ATWS core-damage frequency equal to $1.0x10^{-6}$/yr, and the dominant sequences were found to be:

$T_I(4.2)*C_1(1.0x10^{-5})*\overline{SLC}(.96)*\overline{HADS}(.86)*\overline{P}(.99)*HPLC(6.7x10^{-2})*$
$CDEP(2.13x10^{-1})$ $4.8x10^{-7}$

$T_I(4.9)*C_1(1.0x10^{-5})*SLC(.044)*HADS(.14)*CONT(.224)$ $5.8x10^{-8}$

$T_I(4.2)*C_1(1.0x10^{-5})*SLC(.044)*\overline{HADS}(8.6x10^{-1})*CDEP(2.13x10^{-1})*?(.5)$ $1.7x10^{-7}$

$T_I(4.2)*C_1(1.0x10^{-5})*SLC(.044)*\overline{HADS}(8.6x10^{-1})*VENT(0.9)*CONT(.224)$ $3.1x10^{-7}$

Summary

From the discussion of the most important sequences in the previous sections, it can be seen that the differences between the IDCOR and SARP analyses are principally in the following areas:

(1) Human Actions - In all the three studies, the CDF was mainly determined by the values used for the various probabilities of operator errors. A better perspective of where those values lie is presented in Table A.3, where the conditional probabilities of core damage given a turbine trip or an isolation ATWS are given for the two Peach Bottom studies (in Table A.3 the results of the Shoreham PRA[8] and its review[6] are also presented). From Table A.3, it can be seen that:

- For a turbine trip ATWS, the conditional probabilities of core damage vary from "negligible" (ASEP/SARP) to 0.586 (Shoreham PRA review). The IDCOR value is 0.053.

- For an isolation ATWS, the conditional probabilities of core damage vary from 0.024 (ASEP/SARP) to 0.957 (Shoreham PRA Review). The IDCOR value is 0.261.

- Most of the differences in these CDFs for both ATWS initiators were because of differences in the success criteria which result in substantial differences in human error estimates.

The conditional probabilities of core damage calculated from the Shoreham PRA and its review are also presented in Table A.3 (they will be discussed in Section A.1.3).

(2) Venting - The IDCOR and ASEP/SARP studies gave credit for containment venting and also assumed that even with a failed containment RPV injection with low-pressure systems was possible. If these two "assumptions" are not made, then the ATWS CDF in the IDCOR and ASEP/SARP studies would increase to about 1.2×10^{-5}/yr and 2.3×10^{-6}/yr, respectively.

A.1.2.2 TB (Station Blackout) Sequences

The contribution to core-damage frequency from TB sequences as presented in Tables A.1 and A.4 is:

RSS: 1.0×10^{-7}/yr
IDCOR: 4.5×10^{-7}/yr
ASEP/SARP: 7.0×10^{-6}/yr

Since the RSS analysis assumed that RCIC/HPCI can operate for 27 hours given a blackout, the only contribution to CDF comes from a blackout for more than 30 minutes with independent failure of RCIC and HPCI. Because of this assumption, the RSS analysis was very simplified, and comparisons with IDCOR and ASEP/SARP are not meaningful.

From Table A.4 it can be seen that the main differences between IDCOR and ASEP/SARP come from:

(1) The loss of offsite power (LOOP) initiator frequency used in the ASEP/SARP study was larger than that used in IDCOR by a factor of about 2.6.

(2) A common-mode failure of the dc power system postulated in the ASEP/SARP analysis was responsible for about 70% of the station blackout, core-damage frequency. The IDCOR analysis did not postulate this failure.

(3) About 15% of the station blackout CDF in the ASEP study results from independent failures of two diesel generators (Numbers 2 and 3) because of failures of the diesel generators (DGs) themselves or failures of the emergency service water (ESW). The IDCOR study did not account for these failures because it assumed failure of 3 out of 4 of the DGs compared with the ASEP/SARP assumption of failure of DGs #2 and #3. This

difference in success criteria in the two studies was related to the ESW system dependency on power from DGs #2 or #3.

(4) The IDCOR analysis used a probability equal to 1.7×10^{-3}/demand for common-mode failure of 3 out of 4 DGs, whereas the ASEP/SARP study used a probability equal to 2.35×10^{-4}/demand for common-mode failures of two DGs (#2 and #3).

A.1.2.3 TW Sequences

The contribution to core-damage frequency from TW sequences as presented in Tables A.1 and A.5 is:

RSS: 1.7×10^{-5}/yr
IDCOR: 1.5×10^{-7}/yr
ASEP/SARP: $\sim1.0\times10^{-8}$/yr

The main differences in the three studies are:

(1) Containment Venting and Core Damage Given Containment Failure: The IDCOR and ASEP/SARP analyses, gave credit for containment venting. Venting procedures did not exist at the time of the RSS and thus venting was not accounted for. Human error probabilities (HEPs) equal to 1.0×10^{-2} and 1.0×10^{-3} were used in IDCOR and ASEP/SARP analyses, respectively. The IDCOR analyses also assumed that even if the containment failed there would only be a 10% chance that injection would also be lost and the core would be damaged. If the RSS assumptions (i.e., no containment venting and core damage with containment failure) had been used in all three analyses, the CDF from TW sequences would be given by:

RSS: 1.7×10^{-5}/yr
IDCOR: 2.0×10^{-5}/yr
ASEP/SARP: $\sim7.0\times10^{-6}$/yr

(2) PCS Recovery: The values used for PCS recovery were different in all three studies, as shown in Table A.5. The values used in the IDCOR analysis for failure to recover PCS for DHR ($Q_1 \cdot Q_2$ in Table A.5) were, in general, higher than those used in the other two Peach Bottom analyses.

(3) Human Action: IDCOR was the only analysis that explicitly considered the operator failure to recognize the need for DHR in about 30 hours; a cognitive human error probability equal to 1.0×10^{-6} was used. Note that this was the leading contribution to CDF from TW sequences using IDCOR assumptions.

A.1.2.4 TQUV and TQUX Sequences

The contribution to core-damage frequency from TQUV and TQUX sequences as presented in Tables A.1 and A.6, is:

RSS: 8.4×10^{-7}/yr
IDCOR: 4.1×10^{-8}/yr
ASEP/SARP: 6.8×10^{-8}/yr

The RSS resulted in the higher core-damage frequency because of the success criteria used for the low-pressure injection systems and because at the time of the RSS the RPV required manual depressurization for all transients with failed high-pressure injection (see item 2 below).

The main differences in the three analyses (as given in Table A.6) are:

(1) Control Rod Drive (CRD) System Injection: The SARP analysis was the only one in which credit was given for CRD system pump injection.

(2) ADS: In the IDCOR and ASEP/SARP analysis, the ADS was actuated on a "low-low" level signal alone for 8 minutes. In the RSS, a manual depressurization was required because the actuation of ADS was assumed to require "low" level and "high" drywell pressure signals for 2 minutes.

(3) Low-Pressure Injection: The ASEP/SARP analysis gave credit for injection, after depressurization, with the high-pressure service water (HPSW) system and the IDCOR analysis gave credit for injection with the condensate system.

(4) Loss of dc Bus Initiator: The SARP analysis was the only study that treated this initiator.

A.1.3 Dominant Sequences Comparisons: Peach Bottom and Other BWR Analyses

In this section, an attempt to explain differences in the dominant accident sequences in several Peach Bottom analyses and other BWR PRAs and PRA reviews is made. The most complete BWR analyses that have characteristics similar to Peach Bottom were chosen for this exercise. They are: Limerick PRA Review,[4] Browns Ferry PRA (IREP),[5] and Shoreham PRA Review.[6]

The following sequences for these studies were:

- TC
- TW
- TQUV and TQUX
- TB
- TPQI
- SI
- TPQE

A.1.3.1 TC Sequences

(1) Limerick PRA Review (CDF = 3.7×10^{-6}/yr)

The Limerick plant has an automatic SLCS with 86-gpm capacity and an ARI. With these substantial improvements in the protection system, Limerick would be expected to have a lower TC frequency. But, in that study, the current BWR emergency procedure guidelines were not used.

(2) Browns Ferry - IREP (CDF = 5.5×10^{-5}/yr)

This study does not have a detailed anlaysis of the TC sequences; only the failure of RPS was considered. The dominant TC sequences is:

$T_I(1.7)*RPS(3.0x10^{-5}) = 5.1x10^{-5}/yr$

A comparison with the IDCOR and SARP work for Peach Bottom is also not warranted because of the simplified treatment in the BF-IREP.

(3) Shoreham PRA Review (CDF = $4.5x10^{-5}/yr$).

The major difference in the ATWS CDF between the two Peach Bottom analyses (IDCOR and ASEP/SARP) and the Shoreham PRA review (and also the Shoreham PRA) comes from the HEPs; both the Shoreham PRA review and the Shoreham PRA used higher HEPs than those used in the Peach Bottom studies.

Other important differences were caused by assumptions in the different analyses, i.e.:

- Shoreham PRA and the Shoreham PRA review did not give credit for controlled low-pressure injection; both the SARP analysis and the IDCOR analysis for Peach Bottom did give credit for controlled low-pressure injection.
- Shoreham PRA and the Shoreham PRA Review did not give credit for containment venting.

The effects of these differences can be appreciated by referring to Table A.3 where the conditional probabilities of core damage given an ATWS are presented.

Another difference, of less importance, is that the Shoreham PRA review used higher initiator frequencies and did not account for the power level at the time of the transient.

A.1.3.2 TW Sequences

The contribution to CDF from TW sequences in the Peach Bottom and all other BWR analyses used in this report is presented in Table A.1, and the most important contributions are given in Table A.5.

The most important difference was found to be because of containment venting. Only the IDCOR and SARP analyses for Peach Bottom gave credit for containment venting with failure probabilities equal to $1.0x10^{-2}$ and $1.0x10^{-3}$, respectively. As discussed in Section A.1.2.3, if no credit were given to containment venting, all the results would be between $7.0x10^{-6}$ (ASEP/SARP) and $1.3x10^{-4}$ (Browns Ferry [BF]-IREP). There were two reasons for the high core damage frequency in the BF-IREP: (1) no credit was given to recovery of the PCS for the DHR function and (2) a very high unavailability of the residual heat removal (RHR) system was calculated given no offsite power ($2.9x10^{-2}$).

A.1.3.3 TQUV and TQUX Sequences

The contribution to CDF from these sequences is presented in Tables A.1 and A.6; in Table A.6, the most important sequences are presented. The main differences are:

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(1) Limerick PRA Review (CDF = 6.0×10^{-5}/yr).

The most important differences come from the fact that at the time of the Limerick PRA review, manual depressurization was necessary and a probability of 6.0×10^{-3}/d was used (compared to auto-depressurization in IDCOR and ASEP/SARP), and the unavailability of the high-pressure injection, 8.1×10^{-3}, (compared to 2.0×10^{-3} in IDCOR, and about 4.0×10^{-4} in SARP).

(2) Browns Ferry - IREP (CDF = 5.5×10^{-7}/yr).

The most important difference was the need for manual depressurization (vs. auto-depressurization in IDCOR and ASEP/SARP analysis).

(3) Shoreham PRA Review (CDF = 6.8×10^{-5}/yr).

In the Shoreham PRA review, about 70% of the CDF for these sequences (4.7×10^{-5}/yr) was because of transients not analyzed in the other PRAs (see Table A.6). Some of these sequences were because of particular characteristics of the plant.

For the other 30% of the CDF because of these sequences, the main differences are:

- A higher unavailability of the high-pressure injection system (1.0×10^{-2} vs. 2.0×10^{-3} in IDCOR and about 4.0×10^{-4} in SARP).
- A failure probability of 0.13 was assumed for the high-pressure injection and depressurization functions if offsite power and DGs #1 and #2 were not recovered between 4 and 10 hours. The reason given for this assumption was depletion of batteries.

A.1.3.4 TB Sequences

The contribution to CDF from these sequences is presented in Tables A.1 and A.4. In Table A.4 the most important sequences are shown.

The main differences among the various PRAs are:

(1) Limerick PRA Review (CDF = 3.1×10^{-5}/yr).

- The Limerick PRA review used a LOOP initiator frequency and probability of failure to recover offsite power higher than those used in the IDCOR and ASEP/SARP analyses for Peach Bottom (see Table A.4).
- The Limerick PRA review (and the Limerick PRA) assumed that given a station blackout, HPCI/RCIC can only work for 4 hours (time of battery depletion) if alternate room cooling (ARC) was made available by operator actions, and for 2 hours otherwise. In the IDCOR and SARP analyses, HPCI/RCIC are assumed to work (given a station blackout) for at least 6 hours without any operator action (time of battery depletion).
- The common-mode failure probability of 3 out of 4 DGs used in the Limerick PRA review was about the same as that used in the IDCOR analysis

for Peach Bottom; however, it was much lower than the value used in the ASEP/SARP analysis for two DGs.

(2) Browns Ferry/IREP (CDF = 2.9×10^{-5}/yr).

The major difference between BF-IREP, IDCOR, and SARP analyses for Peach Bottom, comes from the probability of failure of the DGs, which was equal to 2.9×10^{-2} in the BF-IREP. The IREP report did not specify how this value (2.9×10^{-2}) was obtained.

(3) Shoreham PRA Review (CDF = 1.3×10^{-5}/yr).

The Shoreham PRA review (and the PRA itself) presented the most detailed analysis for these sequences. In the review, RCIC/HPCI were assumed to fail at 10 hours after the onset of the transient because of battery depletion. However, for the different time phases used in the analysis (0 to 2 hours, 2 to 4 hours, 4 to 10 hours, and 10 to 24 hours) the failure probability of RCIC/HPCI increased monotonically until 10 hours, beyond which time the probability of failure is unity.

The most important difference between this study and the Peach Bottom analysis was the LOOP initiator frequency together with the failure to recover offsite power.

A.1.3.5 TPQI Sequences

The contributions to CDF from these sequences are presented in Table A.1. The main differences are:

(1) Containment Venting: The SARP and IDCOR analyses for Peach Bottom took credit for containment venting, whereas the others did not; this was the reason why this sequence type did not appear in the IDCOR and SARP analyses for Peach Bottom.

(2) Stuck-Open Safety Relief Valve (SORV): The SARP analysis used a probability of 5.0×10^{-2} for an SORV following a transient. By contrast, the BF PRA used a value of 5.7×10^{-2} for a LOOP initiator and a value of 3.9×10^{-2} for all the other transients. The Shoreham PRA review used a value of 2.0×10^{-2} (for two or more SORVs).

A.1.3.6 TPQE Sequences

The contribution to CDF from these sequences is presented in Table A.1. The only studies that present some contribution from these sequences are:

RSS: 6.0×10^{-8}/yr (inferred from App. V, page V-42, of the RSS)
SARP: 1.8×10^{-7}/yr
BF-IREP: 1.0×10^{-7}/yr

The main reason for this difference came from the probabilities used for SORV (see previous section item 2).

A.1.3.7 Interfacing Systems LOCA

In all the PRAs and the PRA reviews considered in this study, the core damage frequency because of interfacing systems LOCA varied from negligible to 2.0×10^{-7}/yr. The highest interfacing systems LOCA CDF, i.e., 2.0×10^{-7}/yr, which was obtained in the Shoreham PRA review, was mainly because of the fact that the automatic actuation of the ECCS after an interfacing system LOCA would cause flooding of the reactor building. This in turn had the potential to flood the ECCS pumps in a very short time (in some cases less than 10 minutes). Therefore, in the Shoreham PRA and its review, credit was only given to injection with the condensate pumps with makeup to the hotwell. However, other generic studies[9,10] have obtained CDFs that were much higher than those obtained in the PRAs analyzed in this report. Reference 12 has developed the following recommendations, which are intended to "significantly reduce the reactor accident risks" associated with interfacing systems LOCA:

(1) Disable the non-safety-related air operator of the testable isolation check valve on the injection line in the safety systems involved.

(2) Consider the need for leakage testing of the testable isolation check valve before plant startup after each refueling outage or after maintenance, repair, or replacement work on the valve, as an alternative to recommendation 1.

(3) Improve human reliability in maintenance and surveillance testing ties to reduce errors.

(4) Study reducing the frequency of surveillance testing of the isolation barriers of the emergency core cooling systems during power operation.

It should be noted that BNL is currently performing a study on interfacing systems LOCA to provide technical support to the NRC, Reactor and Plant Safety Issues Branch, for the meaningful resolution of this generic issue (GI 104).

A.2 Core-Meltdown Phenomena and Containment Response

In the previous section, important core-meltdown accident sequences were identified in terms of the overall core-melt frequency. In this section, a review of the core-meltdown phenomena and containment response appropriate to these accident sequences is presented. In addition, accident sequences are examined which, although they do not appear to be important to the overall core-melt frequency, may pose a unique or very severe threat to containment integrity. The review will again rely heavily on the IDCOR and SARP analyses which were specifically carried out for the Peach Bottom plant. The review also will take into account other studies pertinent to a BWR-4 with a Mark I containment and, in particular, the results of the Containment Loads Working Group (CLWG) (NUREG-1079)[11] and the Containment Performance Working Group (CPWG) (NUREG-1037).[12]

A typical Mark I containment building is shown in Figure A.1. The Mark I containment volumes are relatively small and rely on water to condense any steam that might be released from the primary system during an accident. Containments of this design are called pressure-suppression containments. Mark I

containments are very effective at condensing steam, but their small volume makes them vulnerable to any combustible and noncondensable gases that would be generated during a severe core-meltdown accident. In fact, the NRC considers Mark I containments to be so vulnerable to combustion that the NRC requires all Mark I containment atmospheres to be continuously inerted with nitrogen during plant operation.

The aim of this section is to identify severe-accident containment loads (pressure/temperature histories) appropriate to the accident sequences identified in Section A.1. These loads are then used to determine the most probable modes of containment failure. These, in turn, identify the potential release paths for fission products to reach the environment. This section, therefore, provides the link between the identification of core-meltdown accident sequences and the determination of fission-product release paths. The results of an assessment of core-meltdown phenomena and containment response is usually expressed in terms of a containment matrix. A containment matrix provides the conditional probabilities of a particular accident sequence resulting in a variety of containment failure modes (or fission-product release paths).

The IDCOR and SARP containment matrices are given in Table A.7. From an inspection of Table A.7, it is clear that the SARP approach included a higher potential for early containment than the IDCOR approach. IDCOR predicted early containment failure only for sequences with loss of CHR. For these sequences, containment failure resulted in loss-of-coolant injection and, hence, core damage. For sequences with core meltdown in an intact containment, IDCOR predicted either no containment failure or failure many hours after reactor vessel failure. In contrast, the SARP analysis yielded a relatively high likelihood of early containment failure for most sequences. In addition, the SARP analysis indicated a higher potential for suppression pool bypass after containment failure and gave less credit for wetwell venting than the IDCOR analysis. Differences in the probabilities in Table A.7 are because of differences in modeling assumptions for core meltdown and containment response in the IDCOR and SARP studies. These differences are discussed in detail in the following sections.

The review of the IDCOR and SARP analyses of core-meltdown phenomena and containment response was greatly assisted by the IDCOR/NRC meetings that were held specifically to identify differences between the approaches adopted by the two groups and to develop a way of resolving these differences. These meetings identified 18 broad NRC/IDCOR issues that highlighted significant differences between the approaches of the two groups. These issues are listed in Table A.8, but they do not all apply to a BWR and some are not related to core-meltdown phenomena and containment response. Out of the 18 issues, 8 have been identified that are appropriate to the subject of this section. Each issue is discussed, in turn, in the following sections. Differences between IDCOR and SARP will be identified and their significance will be indicated.

A.2.1 In-Vessel Hydrogen Generation (NRC/IDCOR Issue 5)

There were (and still are) significant differences between the IDCOR and SARP predictions of hydrogen generation during in-vessel core melting. During the early stages of core heatup and degradation (while the fuel rods are still

in place in the core region), both IDCOR and SARP predicted similar hydrogen generation. However, after the fuel rods and cladding began to melt and relocate into the bottom of the reactor vessel, the SARP analysis indicated more hydrogen generation than the IDCOR analysis.

Hydrogen (H_2) is important to containment loading because it is a combustible and noncondensable gas. The Mark I containments are inerted with nitrogen during plant operation and, consequently, hydrogen combustion (Issue 17) is not a threat to the Mark I containments unless their integrity is lost and oxygen is introduced, thus deinerting the atmosphere. However, Mark I containments are small and, therefore, any significant buildup of noncondensable gases (such as H_2) could threaten containment integrity by overpressure. The larger amount of H_2 generated in-vessel in the SARP analysis resulted in a higher predicted containment pressure before vessel failure than in the IDCOR analysis. BNL staff performed an extensive assessment[13] of in-vessel H_2 generation, particularly with regard to accidents that resulted in core damage but which were terminated by subsequent coolant injection. The results of these calculations indicated the potential for more H_2 generation than predicted by IDCOR. However, both studies allocated a very low probability for overpressure failure of the Mark I containment from in-vessel H_2 generation. The authors concur that there is a low probability of containment failure because of H_2 accumulation before RPV failure and, therefore, the issue does not appear to significantly affect the potential for large fission-product release in Mark I containments. However, this issue is of more importance to other containment designs that are not inerted and is discussed in more detail in other volumes of this report where these other containment designs are addressed.

A.2.2 Core Slump, Core Collapse, and Reactor Vessel Failure (NRC/IDCOR Issue 6)

This is another area in which there were significant differences between the IDCOR and SARP analyses. The importance of these differences to overall risk again depended on plant specifics. Section A.2.1 indicated that the predicted hydrogen generation during core slump was quite different in the IDCOR and SARP analyses but that the impact was not great for the Mark I containment because the atmosphere was inert.

The importance of the core slump and reactor vessel failure models is on how they influence the initial conditions for ex-vessel interactions of the core debris with water or concrete. The IDCOR core slump model assumed that after 20% of the core had melted, it relocated into the bottom of the reactor vessel, which, in turn, rapidly failed because of local penetration failure. Thus, only a relatively small fraction of the core was initially released from the reactor vessel. The remainder of the core melted down over a much longer time period. A similar philosophy has been adopted in the draft NRC staff issue paper on direct heating.[14] This work states that the BWR core support design (which provides individual support for each group of four fuel bundles from the vessel bottom head) is judged to minimize the probability of high-pressure ejection of core debris into the containment. Slumping of relatively small quantities of core debris (because of localized failure of the supports) is anticipated to result in depressurization of the vessel (because of local melt-through) before large quantities of molten core material have collected in the bottom head.

On the other hand, the SARP analysis (with the STCP) assumed total collapse of the core into the bottom of the reactor vessel after 75% of the core was predicted to melt. Thus, all of the core debris was available to be released when the vessel was predicted to fail in the SARP model. The much larger quantity of core materials released from the vessel at the time of vessel failure in the SARP model has important implications for the Mark I containment. If the primary system was at high pressure during core meltdown, then the molten core materials would be ejected under pressure from the reactor vessel when it failed. In Section A.2.4, the phenomenon that could occur when molten core debris is ejected from the reactor vessel under pressure is discussed. Since more core debris was predicted to be ejected (SARP model), the resulting pressure/temperature loads in containment were correspondingly higher. In support of NUREG-1150, SARP has also performed an uncertainty study which examines the range of possible core slump behavior and attaches a low likelihood to the high core-melt fraction slump model.

If the primary system was depressurized during core meltdown, then the core debris would fall under gravity into the region below the reactor vessel after it failed. Obviously, if more core debris was predicted to fall into the pedestal region, then the resulting molten pool would be deeper and there would be a greater potential for the core materials to flow across the drywell floor and reach and fail the steel containment liner. This was identified as a mechanism for early loss of drywell integrity shortly after vessel failure. This potential failure mode was allowed for in the SARP containment event trees but not in the IDCOR analysis (refer to Table A.7). IDCOR has recently submitted an analysis[15] which indicates that the steel liner can survive for several minutes even without drywell sprays. But the key questions are whether the debris is coolable and what is the ultimate depth of debris which contacts the liner.

From the above discussion, it is clear that differences between the IDCOR and SARP models for core slump and vessel failure are significant and do have an important influence on the potential for early containment failure. These differences contribute to the differences in probabilities of early containment failure allocated in the IDCOR and SARP containment event trees (refer to Table A.7).

This is an area of very great phenomenological uncertainty with very little experimental evidence to support either the IDCOR or SARP models. Both models are credible and seem to indicate the range of possible outcomes of a core-meltdown accident. Therefore, given the uncertainty, the possibility of early containment failure should not be ruled out as was done in the IDCOR analysis.

A.2.3 Containment Failure Because of In-Vessel Steam Explosions (Issue 7)

The potential for an in-vessel steam explosion to occur and generate a missile capable of failing containment was investigated by a group of experts and the results were published in NUREG-1116.[16] The conclusion of this expert group was that the occurrence of such an event has a relatively low probability. These results are reflected in the SARP containment event trees. The IDCOR event trees allocate zero probability to an in-vessel steam explosion resulting in containment failure. A very low conditional probability (10^{-4}) of occurrence to this event is supported by the authors.

A.2.4 Direct Heating of Containment (Issue 8)

This was identified in the SARP analysis as an area of significant phenomenological uncertainty related specifically to core meltdown with the primary system at high pressure. If molten core materials were to be ejected from the reactor vessel under pressure, experiments[17] at Sandia National Laboratory (SNL) have indicated that they could form fine aerosols, which might be dispersed into the containment atmosphere and directly heat it. An additional concern was the oxidation of the metallic content of the core debris. These reactions are very exothermic and would add an additional heat load to the containment. Exothermic chemical reactions were less of a concern in Mark I containments because they are inerted (low oxygen content). However, the zirconium-steam reaction could still contribute to containment loading.

The pressure rise in containment because of direct heating is directly proportional to the quantity of core debris dispersed from the reactor vessel. Section A.2.2 noted that the SARP analysis predicted significantly more debris release at vessel failure than the IDCOR analyses predicted. Thus, the potential for early containment failure because of direct heating was considered in the SARP analysis, but it was not considered credible in the IDCOR analysis. The assumption that all the core debris is released at vessel failure (SARP analysis) is clearly conservative (SARP assumption) with respect to containment loading. However, the IDCOR results may be too optimistic considering the lack of supporting large-scale experiments. In addition to the pressure loads imposed by the dispersed core materials, there is the concern that the hot core debris could contact the steel containment shell and rapidly fail it.

Therefore, given the present state of phenomenological uncertainty associated with this issue and the existence of relatively simple mitigative solutions, the following points are offered. There are two ways of potentially mitigating the effects of a high-pressure meltdown. The first is to convert a high-pressure sequence into a low-pressure sequence by ensuring that the automatic depressurization system (ADS) is activated. Note that venting may also be required for some sequences (e.g., station blackout) to ensure that the containment pressure does not increase beyond the relief valve capability. The second way to mitigate a high-pressure meltdown is by drywell sprays. Drywell spray operation may reduce the pressure pulse associated with direct heating, and flooding the drywell floor may impede the dispersal of the core debris and reduce the potential for the core to attack the containment wall. In addition to reducing pressure and cooling the core debris, the drywell sprays will aid in decontaminating the drywell atmosphere and may substantially reduce the released fission products even for cases with drywell failure.

A.2.5 Ex-Vessel Heat Transfer Model From Molten Core to Concrete (Issue 10)

This issue was of concern to Mark I containments because heat transfer from the top of molten core materials (on the drywell floor) directly heats the drywell atmosphere. Thus, differences in heat transfer from the top of the core debris resulted in significant differences in the predicted drywell atmospheric pressures and temperatures. The IDCOR model transferred more heat from the top of the core debris than the SARP model. Thus, IDCOR predicted much higher drywell temperatures than the SARP analyses. However, because IDCOR predicted high heat transfer from the top of the core debris, the

concrete erosion velocities were much lower than the SARP predictions. Lower concrete erosion results in less gases and aerosols released from core-concrete attack and thus lower pressures in containment. Therefore IDCOR predicted containment failure because of high drywell temperatures, whereas SARP predicted containment failure because of high pressure. In addition, IDCOR predicted much longer times to fail the containment because of overtemperature than the times predicted by SARP to fail the containment because of overpressure.

Differences in the predicted drywell pressure/temperature histories influenced the potential for suppression pool bypass (Issue 13A) and containment performance (Issue 15).

A.2.6 Suppression Pool Bypass (Issue 13A)

If the fission products pass through the suppression pool, both IDCOR and SARP predicted significant retention of fission product aerosols in the water. The amount of retention depended on several factors such as submergence, water temperature, aerosol particle size, carrier gas composition, and others. The ability of the Mark I suppression pool to trap aerosol fission products was found to be an important mitigative feature. Thus, any pathways that might open, which would allow the fission products to bypass the pool are very undesirable. The following are possible ways in which the suppression pool may be bypassed:

- loss of drywell isolation
- failure of vacuum breakers between the drywell and wetwell
- failure of drywell penetrations because of high temperature
- structural failure of the drywell because of high pressure
- failure of the drywell wall as a result of contact with molten core materials

Because of the importance of the suppression pool as a mitigative feature, the vulnerability of a Mark I containment to any of the above bypass pathways was carefully assessed. The authors support the low probabilities given in the SARP analysis for loss of drywell isolation and failure of the vacuum breakers. The degradation of the drywell penetrations because of high temperatures in the SARP analysis was based on the work of the CPWG, which had significant BNL input. Failure of the containment steel wall as a result of contact with molten core materials has been identified as a potential failure mode by Greene, Perkins, and Hodge[18] and the revised SARP event trees gave it about a 50% likelihood for station blackout sequences. Given the lack of relevant experiments, the authors were unable to rule out liner melt-through as a potential cause of early containment failure. Thus, Section 4.1 develops criteria to reduce the likelihood of liner melt-through as well as other suppression pool bypass modes.

A.2.7 Containment Performance (Issue 15)

The response of a Mark I containment to severe-accident loads is uncertain. In Section A.2.5, it was noted that IDCOR predicted very high drywell temperatures and thus predicted containment failure because of overtemperature. IDCOR assumed that a relatively small opening would occur which would allow gradual leakage of the drywell atmosphere to the reactor building.

By comparison, the SARP analysis also allowed for the possibility of primary containment failure because of overpressure and assumed an opening large enough to rapidly depressurize the primary containment. In addition, the SARP analysis allowed for degradation of drywell seals because of high temperatures (but lower than the temperature used by IDCOR as its failure criterion). Seal degradation was assumed to result in a gradual leakage from the drywell in the SARP analysis. This was again based on the work of the CPWG.[12]

Differences in containment performance can influence the timing and quantities of fission products released to the reactor building (refer to Section A.3). However, these differences do not lead to major differences in the predicted overall risk as discussed in Section A.4.

A.2.8 Secondary Containment Performance (Issue 16)

The secondary containment (reactor building) surrounds the primary containment and has the potential to trap fission products released during a severe accident. There were differences between the IDCOR and SARP analyses with regard to hydrogen combustion in the secondary containment. These differences resulted in significant differences in the amount of fission products predicted to be retained in the secondary containment. This issue is discussed in more detail in Section A.3.6.

A.3 Fission-Product Release

Section A.2 identified potential containment failure modes or fission-product release paths appropriate to the important core-meltdown accident sequences identified in Section A.1. The aim of this section is to determine the timing and amount of fission products released from the damaged fuel and predict the subsequent reduction of these fission products along the release paths identified in Section A.2. The IDCOR and SARP analyses for the Peach Bottom plant were used as the basis for these calculations.

In order to review differences in approach, the IDCOR and NRC contractor analyses (performed for SARP) are compared in Tables A.9 and A.10 for ATWS and station blackout sequences respectively. The IDCOR methods predicted slightly higher releases of the more volatile fission-product groups (iodine and cesium) whereas the SARP methods predicted much higher releases of the less-volatile fission-product groups (strontium, lanthanum, etc.). The reasons for the different predictions in Tables A.9 and A.10 are complex but were clearly identified during the numerous IDCOR/NRC meetings and they are included in the list of 18 NRC/IDCOR issues in Table A.8. Out of the 18 issues, 6 are pertinent to fission-product release and transport. However, not all of the 6 are major contributors to the differences in Tables A.9 and A.10. Each of the 6 issues is discussed in the following subsections.

A.3.1 Fission-Product Release Before Vessel Failure (Issue 1)

This is one issue that does not contribute significantly to the differences between the IDCOR and SARP analyses in Tables A.9 and A.10. Both studies predicted similar releases of the more volatile fission products during in-vessel core degradation with the exception of tellurium (Te). However, a recent report by IDCOR assessed the impact of Te treatment and modeled similar in-vessel Te releases to the SARP analyses. Differences in the predicted

environmental releases of Te in Tables A.9 and A.10 are, therefore, not because of differences in the in-vessel Te release and retention models, but are due to differences in the amount of retention predicted to occur in the secondary containment (also refer to Section A.3.6).

A.3.2 Fission-Product and Aerosol Retention in the Primary System (Issue 4)

Differences in the initial primary system retention predicted by IDCOR and SARP were again not too significant and differ by less than a factor of two. The important difference between the IDCOR and SARP models was that in the SARP analysis, fission products retained in the primary system at the point of vessel failure were permanently retained, whereas in the IDCOR analysis, revaporization of these fission products after vessel failure was modeled. This is discussed in more detail in Section A.3.4.

A.3.3 Ex-Vessel Fission-Product Release (Issue 9)

There were significant differences between the IDCOR and SARP analyses for fission-product release as a result of core-concrete interactions. The higher releases of the strontium, lanthanum, and cesium groups in Tables A.9 and A.10 in the SARP analyses were because of the modeling of ex-vessel fission-product release. The potential for fission-product release and inert aerosol generation during core-concrete interactions was not modeled in the IDCOR analysis of Peach Bottom.[19] IDCOR argued that by modeling the aerosol generation during core-concrete interactions, the increased aerosol density in containment would increase aerosol agglomeration and settling, thus reducing the predicted environmental release fractions relative to those predicted without this additional aerosol source. This IDCOR position is clearly indefensible.

In addition, the IDCOR predicted core debris temperatures during core-concrete interactions were high and, based on experimental evidence, one would expect the release of some of the refractory fission-product groups at these temperatures. Therefore, the authors believe that IDCOR should calculate the release of the refractory fission products and the associated inert aerosols. The SARP analysis modeled the release of the refractory fission products and the inert aerosols and the environmental release fractions were not calculated to be low (refer to Tables A.9 and A.10).

A.3.4 Revaporization of Fission Products From the Primary System (Issue 11)

Section A.3.2 indicated that revaporization was an area of major difference between the IDCOR and SARP analyses. Recall that SARP does not model revaporization of fission products from the primary system after reactor vessel failure, whereas IDCOR does model this effect. To observe the influence of the different approaches, the distribution of cesium-iodide (CsI) is tracked at various stages of a station blackout sequence in Table A.11.

At the point of vessel failure, both IDCOR and the NRC contractors' analyses performed for SARP indicated significant primary system retention of CsI (100% retention for IDCOR and 85% retention for SARP). Immediately before containment failure, the NRC contractor analysis has 85% of the CsI permanently retained in the primary system, whereas the IDCOR model has revaporized

24% of the CsI from the primary system. At the end of the calculation, the IDCOR model has revaporized 91% of the CsI.

The IDCOR revaporization model means that significantly more of the volatile fission products were predicted to be released to the reactor building than in the NRC contractor approach in which revaporization was not modeled. For the sequence under consideration in Table A.11, the environmental releases were similar despite the major differences discussed above. The reason for the similar environmental releases was because of the much greater retention of fission products in the reactor building predicted in the IDCOR model vs. the SARP model. This is discussed in more detail in Section A.3.6.

A.3.5 Fission-Product Deposition Model in Containment (Issue 12)

This was another issue that does not contribute significantly to the differences between the IDCOR and SARP analyses in Tables A.9 and A.10. Issues 9, 11, and 16 really drive the differences in these two analyses. However, this issue may be of more importance to other containment designs.

A.3.6 Secondary Containment Performance (Issue 16)

Section A.3.4 indicated that secondary containment (reactor building) performance was an area of major difference between the IDCOR and SARP analyses. The extent of the difference can be seen from Table A.11, which shows that IDCOR predicts that only 6% of the CsI entering the reactor building would be released to the environment compared with 40% released in the NRC contractor analysis.

The above differences were significant and were found to be because of several factors. For example, the gas flowrate through the secondary containment was higher in the SARP analysis, which allows less time for aerosol settling. In addition, IDCOR models natural circulation paths in the secondary containment, which further increased the residence times. Such paths were not modeled in the SARP analysis. Finally, SARP calculated several hydrogen burns to occur in the secondary containment, which rapidly blew out the atmosphere of the building. IDCOR calculated more gradual combustion phenomena, which did not result in rapid blowout of the secondary containment atmosphere.

There was uncertainty with regard to how much retention of aerosol fission products would occur in the secondary containment. Even in the SARP analysis, there was the potential for significant retention. However, given the uncertainty, the secondary containment should not be relied on as the only means of mitigating fission-product releases.

A.4 Offsite Consequences

In this section, the potential offsite consequences of the severe accidents described in the previous sections are examined. There is one NRC/IDCOR issue related to offsite consequences, which concerned differences in the assumed evacuation models. Differences in the evacuation model influence the predicted early health effects. The issue was largely resolved and was related to the fraction of the population assumed not to participate in the evacuation.

Table A.12 gives the person-rem calculations performed by IDCOR[20] and in NUREG-1150[14] for several accident sequences and failure modes. This table indicates that if the containment is predicted to fail (either early or late) and the suppression pool is bypassed, then the offsite person-rem predictions are similar (within the range 0.1 to 3×10^7) for the accidents considered. The only time that a significant reduction (to 4×10^5) in person-rem was calculated, was with successful wetwell venting and no pool bypass. These results clearly show that wetwell venting and the prevention of pool bypass are the keys to mitigating the fission-product releases for a Mark I containment.

A.5 Summary and Risk Insights

A.5.1 Core-Damage Profile

As has been observed by others for BWRs, transients rather than LOCAs dominated the core-damage risk profile for the studies examined in Section A.1. Otherwise, there was no consistent pattern of relative ranking of transient sequences among the studies. It is also important to observe that for a given accident sequence in Table A.1, the major contributor to differences in quantitative results between the studies was because of subjective modeling assumptions rather than plant differences or data differences.

For the four BWR plants considered in the six PRAs, the same handful of accident sequences figured prominently in all of the respective core-damage frequency profiles. This suggests that if the probability of this relatively small subset of accident sequences can be minimized, then there is a reasonable expectation that the overall core-damage frequency will be minimized. This principle is used to develop guidelines and criteria to reduce the overall core-damage frequency (Goal 3).

For the RSS Peach Bottom study and the Browns Ferry IREP study, loss of containment heat removal sequences (e.g., TW and TPQI) were important contributors to core melt (about 50%). Although the more recent ASEP and IDCOR studies have reduced these sequences based on operating procedures for venting and alternative injection, criteria have been developed in Section 4.3.2 to ensure that these sequences are not dominant for other BWR Mark I plants.

For the Limerick review, Browns Ferry IREP, and Shoreham review, TQUV/TQUX sequences were important contributors to core melt. Most of this contribution was because of a high failure rate for primary system depressurization. Improved ADS reliability is addressed in Section 4.3.3.

It is important to recognize that the qualitative accident sequence descriptors are rather general and broad and that different hardware and/or operational failures in the various BWR-4, Mark I plants could lead to the same general accident sequence. In order to identify the plant-specific (and often unique) vulnerabilities that contribute to a given general sequence descriptor (e.g., station blackout) in a given plant, a plant-specific examination (such as a failure mode and effects analysis coupled with a fault tree/event tree analysis or an equivalent method) would be needed.

A.5.2 Consequence Analysis

The assessment of core-meltdown phenomena and containment response indicated that the Mark I containment is vulnerable to severe-accident containment loads. Unless mitigative actions are taken, a Mark I containment has the potential to fail a short time (a few hours) after the reactor vessel fails. If containment failure occurs in the drywell, any fission products in the drywell atmosphere could pass to the reactor building without the benefit of suppression pool scrubbing. Because of this vulnerability, the predicted offsite consequences were relatively insensitive to the accident sequence definition. In addition, differences in the IDCOR and SARP assessments of containment response and fission-product release also did not result in major differences in the predicted offsite consequences. The only time that a major reduction in offsite consequences was predicted was with successful wetwell venting and no pool bypass. For station blackout sequences, the dominant failure modes were failure of the drywell shell via contact with core debris or overpressure because of the buildup of noncondensable gases. Thus, both of these failure mechanisms must be reduced substantially to ensure mitigation of fission products (Goal 1).

A.6 References

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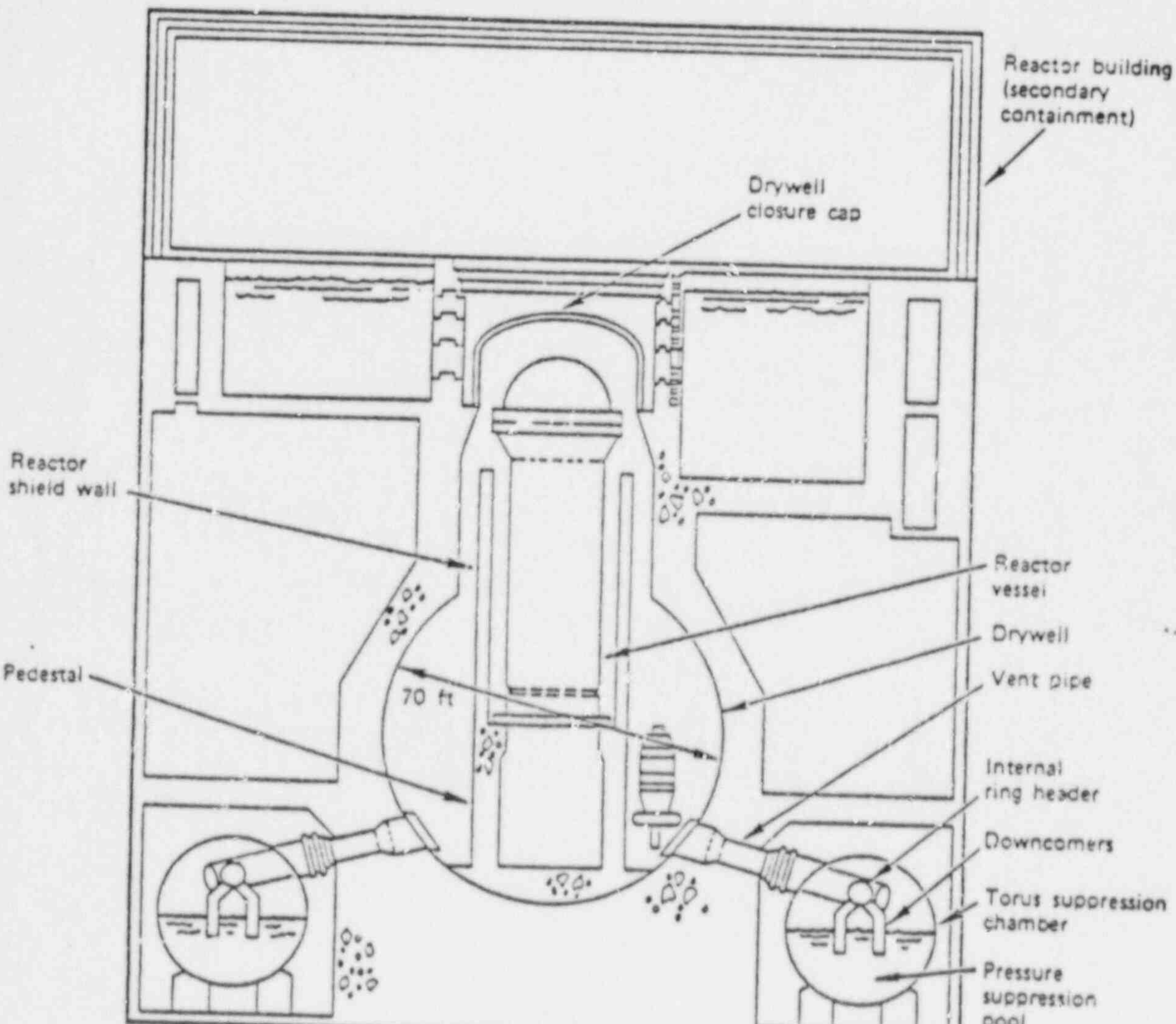

Figure A.1 BWR Mark I containment configuration.

Table A.1 BWR Comparisons: Core-Damage Frequencies

Sequence Type	RSS Peach Bottom	ASEP Peach Bottom	IDCOR Task 21.1 Peach Bottom	Limerick PRA Review	Browns Ferry IREP	Shoreham PRA Review
TW	1.7×10^{-5}	$\sim1.0\times10^{-8}$	1.5×10^{-7}	3.2×10^{-6}	1.0×10^{-4}	9.0×10^{-6} (1.1×10^{-5})
TC	1.3×10^{-5}	1.0×10^{-6}	7.3×10^{-6}	3.7×10^{-6}	5.5×10^{-4}	4.5×10^{-5}
TQUV and TQUX	8.4×10^{-7}	6.8×10^{-8}	4.1×10^{-8}	6.0×10^{-5}	5.5×10^{-7}	5.0×10^{-5} (6.8×10^{-5})
TB	1.0×10^{-7}	7.0×10^{-6}	4.5×10^{-7}	3.1×10^{-5}	2.9×10^{-5}	1.3×10^{-5}
TPQI	---	---	---	9.0×10^{-7}	1.1×10^{-5}	1.7×10^{-7}
TPQE	6.0×10^{-8}	---	---	---	$\sim1.0\times10^{-7}$	---
AE	1.5×10^{-7}	3.2×10^{-8}	1.6×10^{-8}	2.4×10^{-9}	---	3.0×10^{-7}
AI	---	---	---	---	---	---
S_1E	2.0×10^{-7}	7.5×10^{-8}	1.2×10^{-7}	---	---	2.5×10^{-7}
S_2E	5.2×10^{-8}	---	4.2×10^{-9}	---	---	---
S_2I	1.2×10^{-7}	---	---	---	---	---
S_2[illegible]	1.1×10^{-7}	---	---	---	---	---
Total	3.2×10^{-5}	8.2×10^{-6}	8.1×10^{-6}	9.9×10^{-5}	2.0×10^{-4}	1.4×10^{-4}

Table A.2 Summary of Changes Included in the IDCOR Committed Core-Damage Profile for Peach Bottom

Plant Modifications
• Symptom-based emergency procedures including venting, alternate injection sources, ATWS procedures, and using drywell sprays in certain circumstances
• Alternate rod insertion (ARI)
• 86 gpm equivalent manual SLCS
• Level 1 MSIV closure setpoint
• Elimination of "high" drywell pressure permissive for ADS actuation
System Analysis
• New initiating event frequencies
• Increase in operator error contribution to achieve subcriticality
• Lower scram system reliability
• Operator error of cognition considered
• More realistic RSS Case II success criteria
• RHR-HPSW intertie, condensate, and CRD pumps included in some sequences
• Revised treatment of PCS availability and alternative methods of heat removal
• Containment failure not assumed to always result in core damage
• New values for recovery of offsite power and diesel generator reliability and recovery
• Battery depletion considered in station blackout sequence

Table A.3 Conditional Probabilities of Core Damage Given an ATWS in Peach Bottom vs. Shoreham

Analysis	Turbine Trip ATWS[a]	Isolation ATWS[b]
IDCOR Committed		
• High Power	0.067	0.317
• Low Power	0.015	0.160
• Combined	0.053	0.261
ASEP/SARP	Negligible	0.032
Shoreham PRA		
• High Power	0.248	0.893
• Low Power	0.038	0.620
• Combined	0.121	0.831
Shoreham PRA Review		
• All Power Levels	0.586	0.957

[a]Transients in which PCS remains available.
[b]Includes transfers from turbine trip with early closure of MSIVs.

Table A.4 TB Sequences: Comparisons

Study	LOOP T_L	Recover Offsite Power L1	High Pressure Injection U	DGCM	DGREC	RCIC/HPCI In Manual Control	Recover Offsite Power (Given L1) L2	Sequence Core Damage Frequency
RSS -Peach Bottom TB=1.0-7	0.2	2.0-1 (1/2 hr)	2.0-3	1.0-3				8.0-8
IDCOR -Peach Bottom TB=4.[illegible]-7	2.7-2	0.44 (1 hr)		1.7-3	0.72 (6 hrs)	0.5	3.7-2 (9 hrs)	2.7-7
		0.44		1.7-3	0.72	success(0.5)	1.96-2 (12 hrs)	1.4-7
		0.44	2.0-3	1.7-3				4.0-8
ASEP/SARP TB=8.7-6	7.0-2	0.4 (~1 hr)	7.5-5 (CM of dc power system)					5.2-6
			Success	2.3-4 (CM of 2 DGs)	0.5 (6 to 8 hrs)		0.1 (6 to 8 hrs)	3.4-7
		0.4	Success	3.1-5 (SW Single Failure)			0.1	8.8-8
		0.4	Success	1.3-4 (Independent Failure of 2 DGs)	0.6 (6 to 8 hrs)		0.1	2.3-7
		0.4	Success	2.5-4 (1 DG Failure and Other In Maintenance)	0.6		0.1	4.4-7
		0.4	Success	3.3-4 (Combination of Failures in the ESWS and DG Failures)	0.6/0.5		0.1	5.6-7
Limerick PRA Review TB=3.1-5	0.17		$\overline{ARC}$=0.85[a]	1.88-3	0.47 (4-hrs)		3.1-1 (4-hr)	2.1-5
			ARC=[illegible].15	1.88-3	0.66(2-hrs)		4.6-1 (2-hr)	8.0-6
		0.54	8.1-3	1.88-3	0.95(1/2-hr)			1.3-6
Browns Ferry IREP=2.9-5	3.0-2		Success	2.9-2[b]	3.0-2(6 to 8 hrs)			2.6-5
	3.0-2		4.2-2(RCIC)	2.9-2[b]	3.0-2			1.1-6
Shoreham PRA Review TB=1.3-5	0.15	0.37(1/2-hr)	7.[illegible]-1 (RCIC/HPCI not failed at 10 hr but assumed failed after 10 hr)	3.6-3 (2-out-of-3)	2.[illegible]-1 (10-hr)		1.9-1 (10-hr)	6.1-6
			1.1-2	2.3-3(All 3 DGs)				1.4-6

[a]Failure of Alternate Room Cooling.
[b]Combinations of 3 diesels.

Table A.5 TW Sequences: Comparisons

Study	Transient T_i	Cognitive Human Error H	PCS Recovered Early Q_1	PCS Recovered Late Q_2	PCS for DHR $Q_1 Q_2$	Decay Heat Removal (DHR) W	Alternate Decay Heat Removal (Containment Venting) A	Injection Fails After Containment Failure E	Sequence Core Damage Frequency
9SS -Peach Bottom TW=1,7-5	$\sum_i T_i$=1.0				7,0-3	2,3-4			1,6-5
	T_L=0.2				2,0-2	2,3-4			9,2-7
IDCOR -Peach Bottom TW=1,5-7	T_{CV}=0.537								5,3-8
	T_{FW}=0.327	1,0-6						1,0-1	3,3-8
	T_M=0.245								2,5-8
	T_T=6.18	1,0-6	2,0-2						1,2-8
	T_{CV}=0.537		1,0	9,2-2	9,2-2	2,3-4	1,0-2	1,0-1	1,3-8
	T_{FW}=0.327		1,4-1	6,0-3	8,4-4	2,3-4	1,0-2	1,0-1	6,3-11
	T_M=0.245		3,0-1	5,5-2	1,6-2	2,3-4	1,0-2	1,0-1	9,5-10
SARP -Peach Bottom TW=1,6-5[a]									
Limerick PRA Review TW=3,2-6	T_F=1.23		0.61	0.19	1,2-1	9,4-6			1,3-6
	T_L=0.17			7,0-2	7,0-2	4,5-5			5,4-7
	T_T=8.17					7,2-8 (Loss of SW)			5,9-7
Browns Ferry (IREP) TW=1,0-4	T_{NUPCS}<1.7				1.[illegible]	5,7-5			9,7-5
Shoreham PRA Review TW=1,1-5	T_{CV}=0.5			1,2-1	1,2-1	4,4-5			2,5-6
	T_L=0.15			3,0-2	3,0-2	3,1-4			1,4-6
	T_{CV}=0.5	RCIC=0.07		1,2-1	1,2-1	1,1-4			4,2-7
	T_T=8.0		8,2-2	1,3-2	1,1-3	4,4-5			3,5-7
	T_M=0.67		4,5-1	3,0-2	1,3-2	4,4-5			3,7-7
	Loss of SW								2,4-6
	Flooding								2,0-6

[a]Leading cutsets not provided.

Table A.6 TQUV and TQUX Sequences: Comparisons

Study	Transients T_i	FW/PCS Q	High Pressure Injection U	Depressurization X	Low Pressure Injection V	Sequence Core Damage Frequency
RSS	$\sum_i T_i=10$	1.0-2	2.0-3	1.5-[illegible][a]		3.0-7
-Peach Bottom		1.0-2	2.0-3		1.5-3[b]	3.0-7
CDF=8.[illegible]-7	$T_L=0.2$	0.2	2.0-3	1.5-3[a]		1.2-7
			2.[illegible]		1.5-3[b]	1.2-[illegible]
IDCOR	$T_{CV}=0.537$	1.0		2.4-5[d]		3.7-8
-Peach Bottom	$T_{FW}=0.327$	1.4-1	2.0-3[c]			$(\sum TQ)\cdot UX$
CDF=[illegible]	$T_M=0.245$	3.0-1			2.8-6[e]	4.4-9
	$T_I=6.18$	2.0-2				$(\sum TQ)\cdot UV$
ASEP/SAPP -Peach bottom CDF=2.4-7[f]						
Limerick PRA Review	$T_I=1.23$	0.61	8.1-3	6.0-3[a]		3.7-5
CDF=6.0-5	$T_L=0.$		8.1-3	6.0-3		8.3-6
	$T_T=$[illegible]	2.0-2	8.1-3	6.0-3		8.0-6
Browns Ferry (IREP) CDF=[illegible]-7	$T_{NOPCS}=1.7$	1.0	1.8-3	1.8-4[a]		5.5-7
Shoreham PRA	$T_T=8.0$	8.2-2	1.0-2	8.4-4		5.5-6
Review	$T_{CV}=0.5$	1.0	1.0-2	8.[illegible]		4.2-6
CDF=[illegible].8-5	$T_M=0.67$	4.5-1	1.0-2	8.[illegible]		2.5-6
	$M_S=4.3$	3.0-2	1.5-2	8.4-4		1.6-6
	LOOP/Water Level					
	Flooding					1.0-5
	Ref. Leg/CW Cooling					1.8-5
	Loss of [illegible] SW					1.2-5
	Loss of DC					5.2-6
						2.2-6

[a]Manual Depressurization.
[b]Case I WASH-1400.
[c]Unavailability from WASH-1400.
[d]Auto Depressurization.
[e]Case II WASH-1400 (2.8-4) and Condensate Injection (1.0-2).
[f]Leading cutsets not provided.

Table A.7 Comparison of the IDCOR and SARP Containment Matrices

Containment Failure Mode	Station Blackout		ATWS	
	IDCOR	SARP	IDCOR	SARP
No Containment Failure	.72	.06	-	.00
Wetwell Venting or Overpressure	-	.17	.78	.38
Late Temperature-Induced Leak in Drywell	.28	.02	-	.00
Drywell Breach from Melt-Structure Attack	-	.40	-	.03
DW Overpressure Failure	-	.20	.01	.44
Late Pressure-Induced Drywell Failure with Core Quenched	-	-	.12	-
Containment Venting with Loss of Drywell Integrity	-	.15	.09	.15

Table A.8 NRC/IDCOR Issues

Issue	Subject
1	Fission-Product Release Before Vessel Failure
2	Recirculation of Coolant in Reactor Vessel
3	Release Model of Control Rod Materials
4	Fission-Product and Aerosol Retention in the Primary System
5	In-Vessel H_2 Generation
6	Core Slump, Core Collapse, and Reactor Vessel Failure
7	Containment Failure Because of In-Vessel Steam Explosions
8	Direct Heating of Containment
9	Ex-Vessel Fission-Product Release
10	Ex-Vessel Heat Transfer Model From Molten Core to Concrete
11	Revaporization of Fission Products From the Primary System
12	Fission-Product Deposition Model in Containment
13A	Suppression Pool Bypass (Pool Scrubbing)
13B	Retention of Fission Products in Ice Beds
14	Modeling of Emergency Response
15	Containment Performance
16	Secondary Containment Performance
17	Hydrogen Ignition and Burning
18	Essential Equipment Performance

Table A.9 Comparison of IDCOR and SARP Predictions of Fission-Product Release for an ATWS Sequence With No Operator Actions Taken

Event	IDCOR*	NRC Contractors**
Containment Failure (hr)	1.4	1.4
Start of Core Melt (hr)	3.0	2.2
Vessel Failure (hr)	3.9	3.8
Fission-Product Release Fractions***:		
Xe-Kr	1.0	1.0
I-Br	0.1	0.03
Cs-Rb	0.1	0.03
Te-Sb	0.1	0.26
Sr	0.0004	0.49
Ru-Mo	0.001	Neg.
La	--	0.01
Ce	--	0.02
Ba	--	0.39

*IDCOR Technical Report 23.1PB, March 1985.
**NUREG/CR-4624, Vol. 1.
***Fraction of Initial Core Inventory.

Table A.10 Comparison of IDCOR and SARP Predictions of Fission-Product Release for an SBO Sequence

Event		IDCOR*	NRC Contractors**
Loss of Injection	(hr)	6.0	6.0
Start of Core Melt	(hr)	11.4	10.7
Vessel Failure	(hr)	12.0	12.2
Containment Failure	(hr)	18.0	15.2
Fission-Product Release Fractions***:			
Xe-Kr		1.0	1.0
I-Br		0.05	0.012
Cs-Rb		0.05	0.014
Te-Sb		0.06	0.22
Sr		Neg.	0.37
Ru-Mo		0.0001	Neg.
La		--	0.03
Ce		--	0.05
Ba		--	0.28

*IDCOR Technical Report 23.1PB, March 1985.
**NUREG/CR-4624, Vol. 1.
***Fraction of Initial Core Inventory.

Table A.11 Comparison of IDCOR and SARP Predictions of CsI Distribution for an SBO Sequence (Fraction of Initial Core Inventory)

	Events					
	Vessel Failure		Containment Failure		End of Calculation	
Location	IDCOR* (12 hr)	NRC** (12 hr)	IDCOR (18 hr)	NRC (15 hr)	IDCOR (60 hr)	NRC (22 hr)
Reactor Pressure Vessel	1.0	0.74	0.76	0.74	0.09	0.74
Drywell	--	0.12 Melt	0.20	***	--	0.09
Suppression Pool	--	***	0.04	***	0.04	0.14
Reactor Building	--	--	--	--	C.82	0.009
Environment	--	--	--	--	0.05	0.014

*IDCOR Technical Report 23.1PB, March 1985.
**NUREGCR-4624, Vol. 1.
***The location of the remaining fraction of CsI was not reported in the Battelle Columbus Laboratory's report.

Table A.12 Comparison of IDCOR and NUREG-1150 Consequence Results (Person-Rem)

Accident Sequence	Containment Failure Mode	IDCOR	NUREG-1150
ATWS	Wetwell venting no pool bypass	3.5×10^5	--
ATWS	Wetwell venting with late pool bypass	1.0×10^7	--
Station Blackout	Containment failure at RPV failure	--	0.2 to 3.3×10^7
Station Blackout	Containment failure after a few hours	1.3×10^7	0.1 to 1.2×10^7

Appendix B

PLANT FEATURES RESULTING IN LOW PROBABILITIES FOR ACCIDENT SEQUENCES

This appendix examines those BWR-4 plant features which result in insignificant probabilities for many severe-accident sequences. The purpose of this examination is to identify the plant features that produce these low probabilities, and attempt to provide some means, if necessary, to screen other BWR-4 plants to ensure that these same features exist.

In a typical probabilistic risk assessment for a BWR, several thousand severe-accident sequences are examined. Typically, only a few (on the order of 5 to 10) of these sequences are found to contribute significantly (more than a few percent) to the total probability of core damage or risk. The reasons that all other sequences are generally found to be of such low probability are: (1) the frequency of the event that initiates the accident is very low or (2) the protection provided by existing plant systems is highly reliable. In many cases, the low sequence probabilities are the result of both of these reasons acting simultaneously.

All accident sequences examined in a PRA are initiated by one of two general types of events, transient or loss-of-coolant accidents. These two initiators are generally subdivided into several specific classes. It should be noted that there is a special class of transients which can lead to a LOCA by virtue of a stuck-open safety or relief valve. However, this sequence is still initiated by a transient, and the stuck-open valve represents a system (component) failure rather than an initiating event.

As indicated in Appendix A of this report, all PRAs examined consistently showed that transient-initiated sequences dominate core-melt probability. Although the TB and TC sequences are important contributors for the Peach Bottom and Browns Ferry studies, various other transients also show up as significant contributors in some of these studies.

From the preceding discussion, it can be concluded that BWR-4 plants as represented by Browns Ferry and Peach Bottom appear to have reliable protection against loss-of-coolant-initiated accidents. In all cases, LOCA sequences contribute less than 3% to the total CDF, and the maximum probability for any LOCA-initiated core-damage sequence from the four independent studies for Browns Ferry and Peach Bottom is 1.1×10^{-7} (S_1E sequence for the ASEP Peach Bottom study).

The BWR-4 design includes numerous systems for providing emergency core cooling during loss-of-coolant accidents. These systems are shown in Table B.1 (taken from Ref. 1) which illustrates the systems capable of providing adequate cooling as a function of the assumed break size. The table also shows (last column) the decay heat removal success criteria, which are independent of assumed break size or location. As indicated in the second column, the BWR-4 design includes at least three separate systems or combinations of systems capable of providing adequate coolant injection following a LOCA for all break configurations with the exception of the large liquid line break (which, as will be noted later, has a lower estimated probability than other break sizes). Thus, the BWR-4 design includes considerable redundancy and diversity (resulting in high functional reliability) in protection against

LOCA initiators. This design feature has come about as a result of NRC regulations and criteria emphasizing the need for adequate emergency core cooling following LOCAs.

With respect to decay heat removal, Table B.1 indicates that either one or two of four pumps, depending on cooling mode, are required and no diverse system exists to provide this function. However, the system is not required for some time after the LOCA, and failure does not lead immediately to core damage. Thus, opportunity exists for repair to the system, and this is accounted for in some of the PRA studies. For these reasons, loss of decay heat removal following LOCA initiators has not been found to be a significant CDF contributor.

Table B.2 provides an estimate of primary system LOCA break size frequencies as used in the Browns Ferry PRA.[1] Similar values were used in the other BWR assessments. As can be seen from Table B.2, the larger sizes have lower estimated frequencies. The relatively low LOCA frequencies coupled with ECCS and decay heat removal reliability produce insignificant core-damage frequencies from LOCA initiators compared to transient-initiated sequences.

The general ECC system arrangement depicted in Table B.1 for Browns Ferry is expected to exist for all BWR-4 plants since this design is in response to prescriptive NRC requirements for emergency core cooling capability. Documentation by the General Electric Company[2] confirms the design similarity among BWR-4 plants. Thus, it is not expected that any BWR-4 plant exists which would have ECCS design features that would result in a significant estimated core-damage frequency from LOCA initiators. It is, therefore, not considered necessary to require any screening criteria to ensure that BWR-4 plants have adequate ECCS capability for preventing core damage from LOCA initiators.

B.1 References

1. "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," NEDO-24708, General Electric Co., August 1979.

2. "Interim Reliability Evaluation Program: Analysis of the Browns Ferry Unit 1, Nuclear Plant," NUREG/CR-2802, EG&G Idaho, Inc., July 1982.

Table B.1 LOCA Mitigation Success Criteria (BWR-4)

Break Size, Type & Location	Emergency Coolant Injection	Decay Heat Removal
Large, Liquid (0.3 to $4.3\ ft^2$ pump suction)	Two core spray loops and two of four LPCI pumps or Four of four LPCI pumps or One of two core spray loops and two of four LPCI pumps (one LPCI pump per injection loop)	Two of four RHR pumps with associated heat exchangers in torus cooling mode or One of four RHR pumps with associated heat exchangers in shutdown cooling mode
Large, Liquid (0.3 to $4.3\ ft^2$ pump discharge)	Two core spray loops or One of two core spray loops and one of two LPCI pumps on unaffected side	Two of four RHR pumps with associated heat exchangers in torus cooling mode or One of four RHR pumps with associated heat exchangers in shutdown cooling mode
Large, Steam (1.4 to $4.1\ ft^2$)	Two core spray loops or Four of four LPCI pumps or One of two core spray loops and one of four LPCI pumps	Two of four RHR pumps with associated heat exchangers in torus cooling mode or One of four RHR pumps with associated heat exchangers in shutdown cooling mode
Intermediate, Liquid (0.12 to $0.3\ ft^2$)	One of one HPCI pump or Four of six ADS relief valves or One of four LPCI pumps or One of two core spray loops	Two of four RHR pumps with associated heat exchangers in torus cooling mode or One of four RHR pumps with associated heat exchangers in shutdown cooling mode
Intermediate, Steam (pump discharge)	One of one HPCI pump or One of four LPCI pumps or One of two core spray loops	Two of four RHR pumps with associated heat exchangers in torus cooling mode or One of four RHR pumps with associated heat exchangers in shutdown cooling mode
Small, Liquid or Steam (up to $0.12\ ft^2$)	One of one HPCI pump or Four of six ADS relief valves and one of four LPCI pumps or Four of six ADS relief valves and one of two spray loops	Two of four RHR pumps with associated heat exchangers in torus cooling mode or One of four RHR pumps with associated heat exchangers in shutdown cooling mode

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Table B.2 Primary System LOCA Frequencies

Type	Size	Location	Frequency (Per Reactor Year)
Liquid	Large	Suction side	9.9×10^{-6}
		Discharge side	3.9×10^{-5}
Steam	Large	--	5.2×10^{-5}
Liquid	Intermediate	--	9.0×10^{-5}
Steam	Intermediate	--	2.1×10^{-4}
Liquid or steam	Small	--	1.0×10^{-3}

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VOLUME 2

ASSESSMENT OF SEVERE ACCIDENT PREVENTION AND MITIGATION FEATURES:

BWR, MARK II CONTAINMENT DESIGN

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ABSTRACT

Guidelines and criteria have been developed for preventing and mitigating severe accidents in a BWR which has a Mark II containment (BWR Mark II). The guidelines were developed from insights derived from reviews of in-depth risk assessments performed specifically for the Limerick and Shoreham plants and from other relevant studies. Accident sequences that dominate the core-damage frequency and those accident sequences that are of potentially high consequence were identified. Vulnerabilities of the BWR Mark II to severe-accident containment loads were also noted. In addition, those features of a BWR Mark II, which are important for preventing core damage and are available for mitigating fission-product release to the environment were also identified. These guidelines and criteria are issued to provide direction to an analyst examining an individual plant. This direction calls attention to plant features and operator actions and provides deterministic performance measures for assessing those features and actions found to be helpful in reducing the overall risk for Limerick and Shoreham and other Mark II plants. Thus, the guidance is offered as a resource in examining the subject plant to determine if the same, or similar, guidelines will be of value in reducing overall plant risk. These guidelines and criteria are intended to serve solely as guidance.

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NOMENCLATURE

ac	alternating current
ACRS	Advisory Committee on Reactor Safety
ADS	automatic depressurization system
ARI	alternate rod insertion
ASEP	Accident Sequence Evaluation Program
ATWS	anticipated transients without scram
BCL	Battelle Columbus Laboratories
BNL	Brookhaven National Laboratory
BWR	boiling water reactor
C	failure of reactor protection system (RPS)
C_1	mechanical failure to scram
C_2	operator failure to actuate standby liquid control system (SLCS) or to control level with high pressure system, or failure of SLCS
CDF	core-damage frequency
CHR	containment heat removal
CLWG	Containment Loads Working Group
CPWG	Containment Performance Working Group
CST	condensate storage tank
dc	direct current
E	failure of coolant injection
ECC	emergency core cooling
ECCS	emergency core cooling systems
EPG	Emergency Procedure Guidelines
FW	feedwater system
GI	generic issue
HEP	human error probability
HPCI	high-pressure coolant injection system
HPCS	high pressure core spray
HPIS	high-pressure injection systems
IDCOR	Industry Degraded Core Rulemaking Program
IPE	individual plant examination
IREP	Interim Reliability Evaluation Program
ISL	interfacing system LOCA
LER	Licensee Event Report
LGS	Limerick Generating Station
LILCO	Long Island Lighting Company
LLRT	local leak rate testing
LOCA	loss-of-coolant accident
LOOP	loss of offsite power (sometimes denoted by LOSP)
LPCI	low-pressure coolant injection
LPIS	low-pressure injection systems
LWR	light water reactor
MCC	motor control center
MSIV	main steam isolation valve
NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
NRC/RES	U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research
P	one or more stuck open relief valves (SORV)
PCS	power conversion system
PRA	probabilistic risk assessment
PWR	pressurized water reactor

NOMENCLATURE (Cont'd)

Q	failure of feedwater system
RB	reactor building
RCIC	reactor core isolation cooling system
RHR	reactor heat removal system
RPS	reactor protection system
RPV	reactor pressure vessel
RSS	Reactor Safety Study
ry	reactor year
S_1	immediate LOCA
S_2	small LOCA
SARP	Severe Accident Research Program
SARRP	Severe Accident Risk Reduction Program
SBO	station blackout
SLCS	standby liquid control system
SNL	Sandia National Laboratories
SNPS	Shoreham Nuclear Power Station
SORV	stuck open safety relief valve
STCP	Source Term Code Package
T	transient
TB	station blackout sequence (sometimes referred to as SBO)
TC	ATWS
T_I	isolation transients
T_M	manual shutdown
TMI	Three Mile Island
TPQI	transient sequence with SORV failure, feedwater failure and loss of CHR
TQU	transient sequence in which the feedwater and HPI systems fail
TQUX	transient sequence in which the feedwater and HPI systems fail and depressurization does not occur
T_T	turbine trip initiator
TW	loss of CHR
U	failure of high-pressure injection system (HPIS)
USI	unresolved safety issue
V	failure of low-pressure injection system (LPIS)
W	failure of containment heat removal (CHR)
X	failure of reactor pressure vessel (RPV) depressurization

1. EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) has formulated an approach for a systematic safety examination of existing plants to determine whether particular severe accident vulnerabilities are present and what changes are desirable to ensure that there is no undue risk to public health and safety.

The Industry Degraded Core Rulemaking Program (IDCOR) selected four reference plants for detailed analysis: Peach Bottom, Grand Gulf, Sequoyah, and Zion. The IDCOR analyses performed for the reference plants have been documented together with the methodology used for the analyses and the technical basis supporting the methodology.

Parallel with the IDCOR work, the NRC under the Severe Accident Research Program (SARP), performed risk assessments, audit calculations, sensitivity studies, and uncertainty analyses for five plants. The five plants considered by SARP were Peach Bottom, Grand Gulf, Sequoyah, Zion and Surry.

The purpose of this effort is to review all of the IDCOR and SARP analyses performed for the reference plants, understand the reasons for the differences, and then use the experience gained from these reviews for developing guidelines and criteria that identify plant features and operator actions that were found to be important for either preventing or mitigating severe accidents in each plant type. In turn, these guidelines should prove helpful in the systematic safety examination of individual plants.

Three basic objectives or goals for this severe accident program apply equally to all plant types:

- Goal 1: Mitigate fission-product releases.
- Goal 2: Control the frequency of high-consequence sequences.
- Goal 3: Reduce high core-damage frequency.

The aim was, therefore, to develop detailed guidelines and criteria that could be used to achieve these goals during the examination of individual plants.

"Guidelines," as used in this report, identify those plant features and operator actions that were found to be important to either preventing or mitigating severe accidents in the reference plant studies. These guidelines provide a list of plant features and operator actions that the utilities can use as part of each individual plant examination (IPE). It is not the intent of this report to specify a set of improvements for either the reference plant or for any other plant which would be sufficient to achieve a certain level of safety. Instead, the guidelines indicate potential improvements in various areas of plant design and operation of which each utility should be aware when conducting its IPE and making decisions on plant improvements. The intent is to provide guidance to the analyst performing an IPE, as to the plant features and operator actions which were found to reduce overall risk. It is prudent to check whether potential improvements identified in studies of other similar plants can be of help in improving overall plant performance. The guidelines contained in this report, therefore, complement the IPEs.

"Criteria," as used in this report, are the attributes which have been identified as important to assess the performance of plant features and operator actions identified in the guidelines. The criteria provide deterministic (as opposed to probabilistic) performance measures which are judged to be helpful to implement the concepts contained in the guidelines. When a decision is made to provide an item or an action specified in a guideline, the utility should address a set of questions relating to the design, operation and availability of the needed equipment and the training of operators. For the example of containment-venting guidance, the capacity of the venting system, the selection of setpoints to initiate venting, the availability of applicable procedures and the accessibility of certain valves by operators should be addressed. The criteria on containment venting provide helpful information in assessing venting capability in each individual plant.

Based on an extensive review of prior severe accident investigations, the authors have provided a set of guidelines and associated criteria which can be used to assess the capability of individual boiling water reactor (BWR), Mark II plants to cope with severe accidents. Although much of the work is based on probabilistic risk assessments (PRAs), the guidelines and criteria are deterministic in nature. That is the criteria describe specific features of key systems and operational procedures which have been found helpful in reducing the likelihood of severe accidents. The guidelines and criteria take into account detailed severe accident experiments and analyses performed by the NRC/RES, the nuclear industry and foreign govenments.

The following sections present the insights gained from reviewing the PRAs. None of the four IDCOR reference plants or the five plants considered by SARP have a Mark II containment. However, PRAs have been performed for two BWRs with Mark II containments, namely, Limerick[1] and Shoreham[2] and these PRAs have been reviewed in detail by Brookhaven National Laboratory (BNL)[3,4] and the NRC staff.[5]

1.1 Core-Damage Profile

Transients rather than loss-of-coolant accidents (LOCAs) dominated the core-damage risk profile for the BWR Mark II PRAs reviewed. In addition there appeared to be no consistent pattern of relative ranking of transient sequences among the studies. It is also important to observe that for a given accident sequence, the major contributor to differences in quantitative results between the studies was because of subjective modeling assumptions rather than to plant differences or data differences. For the two Mark II plants considered in the five studies, several of the accident sequences figured prominently in all of the respective core-damage frequency (CDF) profiles.

For the Limerick Generating Station (LGS) and the Shoreham Nuclear Power Station (SNPS) PRA reviews, accident sequences with failure of high pressure injection were found to be important contributors to CDF. Most of this contribution was attributable to a high failure rate for primary system depressurization. Therefore, criteria for improved automatic depressurization system (ADS) operability have been developed.

The Peach Bottom studies done by SARP and IDCOR indicated that station blackout (SBO) and anticipated transients without scram (ATWS) are the

dominant core-damage sequences for Peach Bottom. Both studies calculated a total CDF approaching 10^{-5} per reactor year. Both the LGS and the SNPS studies confirmed that SBO and ATWS sequences were major contributors to the Shoreham CDF. Thus, guidelines have also been developed for SBO and ATWS sequences.

For the Reactor Safety Study (RSS)[6] of a BWR Mark I (Peach Bottom) and the Interim Reliability Evaluation Program (IREP) study[7] (which used the Browns Ferry plant) loss of containment heat removal sequences were important contributors to CDF (about 50%). The more recent studies for Peach Bottom as well as Limerick and Shoreham have reduced the CDF because of these sequences based on operating procedures that include venting and alternate injection. Criteria have been developed to ensure that these sequences are not dominant for other BWR Mark II plants.

1.2 Consequence Analysis

The assessment of core-meltdown phenomena and containment response in PRAs reviewed indicate that the Mark II containment is vulnerable to severe accident containment loads. Unless mitigative actions are taken a Mark II containment has the potential to fail shortly (in a few hours) after the core debris melts through the reactor pressure vessel. If containment failure occurs in the drywell, any fission products in the drywell atmosphere could pass to the reactor building without the benefit of suppression pool scrubbing. If the containment failure occurs in the wetwell below the level of the suppression pool, there is potential for loss of the pool and resulting loss of scrubbing.

1.3 Guidelines and Criteria

Each guideline is provided with a detailed list of criteria which provide helpful information to assess the performance of plant features and operator actions identified in the guidelines.

1.3.1 Mitigate Fission-Product Releases

The assessment of core-meltdown phenomena and containment response indicates that Mark II containments are vulnerable to severe accident containment loads because of the relatively small volume. Unless mitigative actions are taken, a Mark II containment has the potential to fail shortly after reactor pressure vessel failure. For this reason the authors developed Guidelines 1 and 2, given in Table 1.1, which have the potential to mitigate the consequences of a severe accident.

Guideline 1 - Maintain Containment Integrity

Mark II containments are very effective at condensing steam, but their small volume makes them vulnerable to any combustible and noncondensable gases that would be generated during a severe core-meltdown accident. The impact of hydrogen burning was not significant for the Mark II containment because the atmosphere is inerted. However, all of the PRAs reviewed predicted that the accumulation, of noncondensable gases, released from core-concrete interaction, will rapidly overpressurize the Mark II containment. In addition, the studies predicted high temperatures in the drywell which could cause drywell

seal degradation. The authors therefore developed Guideline 1.A on containment venting (relating to preventing overpressure failure) and Guideline 2.B (see Table 1.1) on containment spray (relating to preventing failure because of high temperature).

<u>Guideline 2</u> - Maintain Suppression Pool Effectiveness

The ability of the Mark II suppression pool to trap aerosol fission products is an important mitigative feature since it has the potential to significantly reduce offsite consequences. Thus, any pathways that might open, which would allow the fission products to bypass the pool, are undesirable. Therefore, Guideline 2 and the associated criteria are provided to help prevent fission products from bypassing the suppression pool.

1.3.2 Control the Frequency of High-Consequence Sequences

For some types of severe accidents it is difficult to ensure that large-fission-product releases do not occur. Therefore guidelines and criteria have been develped to ensure control of the frequency of high-consequence sequences (Goal 2).

<u>Guideline 3</u> - Interfacing Systems LOCA

In general, BWR Mark II PRAs have found the interfacing systems LOCA to be extremely unlikely. However, the possibility of high releases make it important to ensure that the frequency of these events is kept very low at all Mark II plants.

<u>Guideline 4</u> - Anticipated Transients Without Scram

Anticipated transients without scram (ATWS) have been found to be important contributors to risk for many BWRs. The NRC promulgated the ATWS rule to reduce the frequency of ATWS events. The criteria developed for this guideline emphasize recent insights to the importance of correct emergency procedures and operator training in recovering from an ATWS event.

1.3.3 Reduce High Core-Damage Frequency Sequences

The review of BWR Mark II PRAs has identified a number of potentially important contributors to the CDF. Therefore guidelines and criteria have been developed to reduce the potential for high CDF (Goal 3).

<u>Guideline 5</u> - Station Blackout

For accidents involving the loss of offsite power and onsite emergency power, the NRC recommends examining the proposed station blackout (SBO) rule for applicability. The criteria associated with Guideline 5 are intended to emphasize the need to search for plant specific features and potential common cause failures which could disable systems required to work during an SBO. For individual plants which are found to have a vulnerability to SBO, the criteria highlight the importance of proper emergency procedures and operator training in recovering from an SBO event.

Guideline 6 - Loss of Containment Heat Removal

Accident sequences involving loss of containment heat removal (CHR) were found to be quite important in the earlier PRA studies. In WASH-1400, loss of CHR sequences accounted for 53% of the calculated CDF. In the Browns Ferry[7] IREP study, these sequences similarly accounted for 50% of the calculated CDF. However, the most recent BWR studies by IDCOR and ASEP/SARP (as well as the LGS and SNPS PRAs), show a two-and-three-order-of-magnitude reduction, respectively, in the quantification of these sequences. Therefore, Guideline 6 and the associated criteria have been developed to generically address the mechanisms already effectively employed at the LGS and SNPS to reduce the frequency of loss of CHR sequences.

Guideline 7 - Reactor Pressure Vessel (RPV) Depressurization Performance

One of the insights gained from the existing BWR PRAs is the importance of the ADS in mitigating loss of high pressure injection sequences by allowing the low-pressure injection (LPIS) systems to operate effectively. Specific criteria have been developed to ensure that the likelihood of loss of injection accident sequences occurring is low.

Guideline 8 - Support System Interdependencies

Although the importance is difficult to quantify, one of the insights developed in most risk assessment studies is the importance of support system interdependencies. For example, a draft of the Severe Accident Risk Reduction Program (SARRP) Peach Bottom study indicated that loss of all service water was a dominant contributor to core damage. The recent revision to the sequence studies has reduced it to one percent of the overall core damage. In order to ensure that support system vulnerabilities do not cause unacceptably high CDF for other BWR Mark II plants, the authors have developed this guideline and associated criteria to help assess any weaknesses of the support systems.

Guideline 9 - Flooding Within Reactor Building

Flooding of the reactor building has been found to be a significant contributor to core damage in only one Mark II plant. This guideline and associated criteria were developed for all Mark II plants to assess the potential for flooding of safety related equipment.

1.4 Using the Guidelines and Criteria

Numerous investigations, including PRAs, have been performed for the reference plants and similar plants by both the NRC and the nuclear power industry. The insights gained from many of the studies have been used in developing the guidelines and criteria contained in this report (including the other volumes of this report). These guidelines and criteria are issued to provide guidance to the analyst performing an IPE. This guidance is in the form of plant features, operator actions and the criteria for assessing those features and actions found to be helpful in reducing the overall risk for Limerick and Shoreham and other Mark II plants. Thus, the guidance is given to provide a resource in examining the subject plant to determine if the same, or

similar, guidelines will be of value in reducing overall plant risk. These guidelines and criteria are intended to be used solely as guidance.

1.5 References for Section 1

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6. "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, NUREG/75-014, October 1975.

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Table 1.1 Guidelines for Preventing and Mitigating Severe Accidents in a BWR with a Mark II Containment

Guideline	Description
	Mitigate Fission-Product Releases:
1	Maintain Containment Integrity
1.A.	Provide Wetwell Venting
2	Maintain Suppression Pool Effectiveness
2.A.	Prevent Suppression Pool Bypass
2.B.	Provide Drywell Spray
	Control the Frequency of High-Consequence Sequences:
3	Interfacing Systems LOCA
3.A.	Prevent Overpressurization of Low Pressure Systems
4	Anticipated Transients Without Scram (ATWS)
4.A.	Provide Operator Response During ATWS
	Reduce High Core-Damage Frequency Sequences:
5	Station Blackout
5.A.	Provide Reactor Pressure Vessel Injection
6	Loss of Containment Heat Removal
6.A.	Provide Long-Term Emergency Core Cooling
7	Reactor Pressure Vessel (RPV) Depressurization Performance
7.A.	Provide RPV Depressurization
8	Support System Interdependencies
8.A.	Examine Support System Interdependencies
9	Flooding Within Reactor Building
9.A.	Prevent and Mitigate Reactor Building Flooding

2. INTRODUCTION

2.1 Background

The U.S. Nuclear Regulatory Commission (NRC) has formulated an approach for a systematic safety examination of existing plants to determine whether particular severe accident vulnerabilities are present and what changes are desirable to ensure that there is no undue risk to public health and safety.

The Industry Degraded Core Rulemaking Program (IDCOR) selected four reference plants for detailed analysis, namely:

- Peach Bottom (a BWR with a Mark I containment)
- Grand Gulf (a BWR with a Mark III containment)
- Zion (a PWR with a large dry containment)
- Sequoyah (a PWR with an ice condenser containment)

The IDCOR analyses performed for the above reference plants have been documented together with the methodology used for the analyses and the technical basis supporting the methodology.

Parallel with the IDCOR work, the NRC under the Severe Accident Research Program (SARP), performed risk assessments, audit calculations, sensitivity studies, and uncertainty analyses for five plants. The five plants considered include the above four IDCOR reference plants, and, in addition

- Surry (a PWR with a subatmospheric containment)

The purpose of this effort is to review all of the IDCOR and SARP analyses performed for the reference plants, understand the reasons for the differences, and then use the experience gained from these reviews for developing guidelines and criteria that identify plant features and operator actions found to be important for either preventing or mitigating severe accidents in each plant type. In turn, these guidelines should be helpful in the systematic safety examination of individual plants.

The first BWR plant reviewed was Peach Bottom, which is a BWR-4 with a Mark I containment. The IDCOR Peach Bottom analysis[1] was documented in March 1985 and was supplemented by additional sensitivity studies in July 1985. The SARP Peach Bottom reports[2-4] were reviewed in draft form during 1986. These reports were published early in 1987 and were summarized in the "Reactor Risk Reference Document" (NUREG-1150),[5] which was published for comment in February 1987. The experience gained from the review of these Peach Bottom studies along with other BWR PRA studies (namely, Limerick, Shoreham and Browns Ferry) was used to generate the guidelines and criteria which are the subject of Volume 1 of this report.

This volume builds on the experience gained during our Peach Bottom work and deals with severe accidents in a BWR-4 or BWR-5 plant with a Mark II containment (BWR Mark II). BWR-4 or BWR-5 plants with a Mark II containment have many similarities to BWR plants with Mark I containments, although significant differences also exist. None of the four IDCOR reference plants or the five plants considered by SARP have a Mark II containment. However, PRAs[6,7] have been performed for two plants with a Mark II containment namely the: Limerick

Generating Station (LGS) and Shoreham Nuclear Power Station (SNPS). The experience gained from the reviews[8,9] of these studies as well as the insights gained from the Mark I studies was used to generate the guidelines and criteria, which are the subject of this draft report.

2.2 Objectives

Three basic objectives or goals for this severe accident program apply equally to all plant types:

- Goal 1: Mitigate fission-product releases.
- Goal 2: Control the frequency of high-consequence sequences.
- Goal 3: Reduce high core-damage frequency.

The aim was, therefore, to develop detailed guidelines and criteria that could be used to achieve these goals during the examination of individual plants. The guidelines and criteria are given in the sections that follow.

2.2.1 Guidelines

"Guidelines," as used in this report, identify those plant features and operator actions that were found to be important to either preventing or mitigating severe accidents in the reference plant studies. These guidelines provide a list of plant features and operator actions that the utilities can use as part of each individual plant examination (IPE). It is not the intent of this report to specify a set of improvements for either the reference plant or for any other plant which would be sufficient to achieve a certain level of safety. Instead, the guidelines indicate potential improvements in various areas of plant design and operation of which each utility should be aware when conducting its IPE and making decisions on plant improvements. The intent is to provide guidance to the analyst performing an IPE, as to the plant features and operator actions which were found to reduce overall risk. It is prudent to check whether potential improvements identified in studies of other similar plants can be of help in improving overall plant performance. The guidelines contained in this report, therefore, complement the IPEs.

The three objectives or goals were noted as applying equally to all plant types. Although the goals are independent of plant type, the guidelines that are needed to achieve the goals are plant dependent. In general terms, Goal 1 implies that there should be effective means of mitigating the fission product releases for the broad classes of accident sequences which dominate the core-damage frequency. Therefore, these dominant accident sequences have to be determined and those plant features and operator actions that are available to mitigate the release of fission products have to be identified. Only then can detailed guidelines be developed to ensure that these dominant accident sequences can be mitigated.

There may be accident sequences for which a specific plant will have substantial fission-product releases (e.g., containment bypass sequences). Thus, for such sequences Goal 1 may be difficult to achieve. Therefore, all reasonable steps are identified which could reduce the frequency of these potentially high consequence sequences (namely Goal 2). Again, the accident sequences have to be identified and plant vulnerabilities and/or operator actions that lead to core damage for these sequences also have to be identified.

Detailed guidelines can then be developed which will aid in assessing an individual plant's capability to prevent these sequences from occurring.

It is also desirable to ensure that the overall core-damage frequency is low (namely Goal 3). Again, the dominant accident sequences have to be found so that detailed guidelines can be developed to reduce the frequency of these sequences, if necessary.

In general, the following screening process was used to determine whether or not to develop a particular guideline:

- any accident sequence with a core-damage frequency greater than 10^{-6} per reactor year
- any sequence that contributed to more than 5% of the total core-damage frequency
- any event that caused a conditional probability of early containment failure greater than 0.1
- any sequence that resulted in containment bypass with a frequency greater than 10^{-7} per reactor year
- any sequence that was judged to be uniquely important (example, very severe consequences)

This screening process led to the development of guidelines that can be used in the systematic safety examination of other BWRs with Mark II containments. For example, the guideline for venting of the wetwell was identified as an item that would help to achieve Goal 1 (namely, to mitigate fission-product release) for the BWR Mark II reference plant. Therefore, in the safety examination of other BWRs with Mark II containments, the need for wetwell venting would have to be carefully assessed.

The development of a particular guideline for the BWR Mark II reference plant does not imply that this plant or any of the other plants in this category need to conform to this guideline. It simply means that analyses have indicated that this particular guideline has the potential to significantly reduce risk. Thus, the guidance is given to provide a resource in examining the subject plant to determine whether the same or similar guidelines will be of value in reducing overall plant risk. Whether or not the guideline is useful or needed in a particular BWR with a Mark II containment depends on plant-specific details and is beyond the scope of this report and is therefore not addressed here.

2.2.2 Criteria

"Criteria," as used in this report, are the attributes which have been identified as important to assess the performance of plant features and operator actions identified in the guidelines. The criteria provide deterministic (as opposed to probabilistic) performance measures which are judged to be helpful to implement the concepts contained in the guidelines. When a decision is made to provide an item or an action specified in a guideline, the utility should address a set of questions relating to the design, operation

and availability of the needed equipment and the training of operators. For the example of containment-venting guidance, the capacity of the venting system, the selection of setpoints to initiate venting, the availability of applicable procedures and the accessibility of certain valves by operators should be addressed. The criteria on containment venting provide helpful information in assessing venting capability in each individual plant.

The criteria address the general issues of (1) survivability of equipment (i.e., whenever credit is given for a system or a component to mitigate the accident, the ability of the equipment to function under environmental and fluid dynamic loads associated with severe accident sequences must be taken into account), (2) equipment capabilities, capacities, and duration of operability, (3) accessibility of equipment, (4) availability of support systems, (5) identification of necessary components, (6) identification of important operator actions, and (7) identification of parameters for initiation of mitigating systems and operator actions.

2.3 Organization of the Report

This report describes detailed guidelines and criteria for preventing and mitigating severe accidents in BWRs that have a Mark II containment. It is one of a series of five volumes that deal with guidelines and criteria for several different reactor and containment types. Other volumes in the series are given below:

Volume 1: BWRs with Mark I Containments
Volume 3: BWRs with Mark III Containments
Volume 4: PWRs with Large Volume Containments
Volume 5: PWRs with Ice Condenser Containments.

Appendix A of this volume contains a review of the PRAs that were performed for BWRs with Mark II containments along with other pertinent studies. The insights gained from these studies lead to the identification of the strengths and vulnerabilities of a BWR with a Mark II containment. In Section 3, the three basic goals of the program are related to the relevant design features and operating characteristics of a BWR with a Mark II containment. The guidelines necessary to achieve the three goals are therefore initially developed in Section 3. In Section 4, the guidelines are restated and detailed criteria are developed for each guideline.

2.4 References for Section 2

1. "Peach Bottom Atomic Power Station-Integrated Containment Analyses," IDCOR Technical Report T23.1PB, March 1985.

2. A. M. Kolaczkowski et al., "Analysis of Core Damage Frequency from Internal Events: Peach Bottom, Unit 2," Sandia National Laboratories, NUREG/CR-4550, Volume 4, October 1986.

3. C. N. Amos et al., "Containment Event Analysis for Postulated Severe Accidents: Peach Bottom Atomic Power Station, Unit 2," Sandia National Laboratories, NUREG/CR-4700, Volume 3, Draft Report for Comment, May 1987.

4. C. N. Amos et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Peach Bottom, Unit 2," Sandia National Laboratories, NUREG/CR-4551, Volume 3, Draft for Comment, April 1987.

5. "Reactor Risk Reference Document," U.S. Nuclear Regulatory Commission, NUREG-1150, Draft for Comment, February 1987.

6. "Probabilistic Risk Assessment - Limerick Generating Station," Philadelphia Electric Co., 1982.

7. "Probabilistic Risk Assessment - Shoreham Nuclear Power Station," Long Island Lighting Company, June 1983.

8. I. A. Papazoglou et al., "A Review of the Limerick Generating Station Probabilistic Risk Assessment," Brookhaven National Laboratory, NUREG/CR-3028, February 1983.

9. D. Ilberg et al., "A Review of the Shoreham Nuclear Power Station Probabilistic Risk Assessment (Internal Events and Core Damage Frequency)," Brookhaven National Laboratory, NUREG/CR-4050, June 1985.

3. DEFINITION OF GOALS AND RELEVANT BWR MARK II FEATURES

In Section 2 of this report, the concept of three basic objectives or goals for this severe accident program was introduced. The concept applied equally to all plant types. In this section, the three goals are related to the relevant design features and operating characteristics of a BWR with a Mark II containment for the accident sequences and containment failure modes found to be important in Appendix A. This includes consideration of both favorable and unfavorable severe accident attributes.

Screening criteria have been used to identify those sequences that need to be addressed by severe accident guidelines for each goal. Specifically;

- For Goal 1 (Mitigate fission-product releases), all sequences have been examined which represent 5% of the core-melt frequency or are estimated to occur more often than 10^{-6} per reactor year and result in a conditional probability of early containment failure greater than 0.1.

- For Goal 2 (Control the frequency of high-consequence sequences), all sequences have been examined which result in pool bypass and are estimated to occur more often than 10^{-7} per reactor year.

- For Goal 3 (Reduce high core-damage frequency sequences), all sequences have been examined which "have the potential to occur" more frequently than 10^{-6} per reactor year. Note that this screening criterion has been used to identify potential vulnerabilities from risk assessment insights which do not necessarily apply to the LGS and SNPS, but may apply to other Mark II plants.

This section provides the link between the goals (developed in Section 2) and the guidelines (developed in Section 4) that may be used to assess the capability of specific plants to meet these goals. This section is organized into three subsections, which correspond to the three goals.

3.1 Mitigate Fission-Product Releases

Goal 1 requires that there shall be effective means of mitigating the fission-product releases for the broad classes of accident sequences that may lead to core damage in a BWR with a Mark II containment. In Appendix A the most important contributors to the core-damage frequency (CDF) were found to be SBO and ATWS sequences. Several studies indicate that transients with loss of injection into the reactor pressure vessel (RPV) are also potentially important contributors. Other transients and loss-of-coolant accidents (LOCAs) may also contribute to CDF. Two specific accident sequences for which mitigation by a Mark I containment is ineffective (refer to Volume 1) are also applicable to Mark II's. These specific sequences are discussed further in Section 3.2, which attempts to determine how the frequency of these unmitigated sequences can be reduced. This section concentrates on the broad classes of accident sequences for which plant features provide significant mitigating fission-product release. In the following sections both the favorable and unfavorable severe accident attributes of the Mark II containment are identified. This discussion in turn leads to the development in Section 3.1.3 of two guidelines that are related to Goal 1.

3.1.1 Plant Vulnerabilities

The Mark II containment is a small-volume, pressure-suppression design. The suppression pool is available to condense steam released from the primary system during an accident. However, the small volume of the Mark II containment makes it vulnerable to pressure and/or temperature increases because of the noncondensable gases and heat released during a core-meltdown accident. There are differences among Mark II containment analyses regarding the estimates of how long it will take to pressurize a Mark II containment to its ultimate capacity after the core debris has failed the reactor vessel (and is interacting with concrete); but most studies concluded that the containment will eventually fail. Therefore, unless mitigative actions are taken, a Mark II containment will fail because of overpressure or overtemperature within a few hours of RPV failure. If containment failure occurs in the drywell, any fission products in the drywell atmosphere could pass to the reactor building (and ultimately to the environment) without the benefit of suppression pool scrubbing. Note that suppression pool scrubbing is an important mitigative feature of a Mark II containment (refer to Section 3.1.2).

An inspection of the Mark II containment configuration (see Figure A.1) shows that the pedestal below the RPV would tend to confine the core debris after a core-meltdown accident. Extensive core-concrete interactions would be expected to occur. There are differences between the IDCOR and SARP analyses related to how high the core debris temperature will remain during these interactions and to the quantities of the less volatile fission products that will be released. However, at this time the possibility of the core debris remaining hot and releasing significant quantities of fission products has not been ruled out. In addition, some Mark II designs have downcomers within the pedestal which would allow much of the core debris to be channeled into the pool. The pedestal designs differ so much among different Mark II plants that the progress of the core debris, once it reaches the pedestal region, cannot be predicted in a generic manner (see Figure A.2). For some Mark II's the debris will flow out of the pedestal region onto the drywell floor. Unlike the Mark I situation, penetration of the drywell shell by core debris in this scenario can be virtually ruled out, since the drywell floor of Mark II's is significantly larger, downcomers allow the debris to fall into the wetwell before reaching the shell, and the shell itself is usually concrete not steel. However, debris accumulation on the drywell floor, and the resulting transport into the suppression pool, could attack and fail parts of the drywell to wetwell gas space boundary. Therefore the wetwell pool scrubbing potential could be lost for accident sequences where the wetwell air space is vented or failed. For some Mark II designs the debris will remain in the pedestal cavity between drywell and wetwell out of reach of drywell sprays or the suppression pool. Sufficient core-concrete interaction may open a path from the debris to the wetwell airspace via the pedestal wall again allowing some fission products to bypass suppression pool filtering.

In the sections that follow, suppression pool scrubbing is noted as an effective mitigative feature for the Mark II containment provided all of the fission products pass through the pool. It is, therefore, important to ensure that paths do not open which would allow the fission products to bypass the suppression pool. The vacuum breakers between the wetwell and drywell would create a path that bypasses the suppression pool if they fail open. In addition, the various drywell penetration seals could be degraded at high

temperatures and pressures. Failure of these seals would also open up paths which would bypass the suppression pool. If the main steam isolation valves (MSIVs) fail to close, another suppression pool bypass path would exist.

Although the Mark II containments appear to be vulnerable to severe accident containment loads, they have several very important mitigative features, which are discussed in the section that follows.

3.1.2 Mitigating Features

The suppression pool in a Mark II containment is a very effective mechanism for trapping any fission-product aerosols that might pass through it. Thus, to a large extent the suppression pool has the potential to compensate for the vulnerabilities identified above (in Section 3.1.1). For example, overpressure failure of the containment (and perhaps loss of drywell integrity) can be prevented by venting the wetwell. With venting of the wetwell atmosphere, containment integrity is lost but the containment function (retention of the fission products in the pool) is maintained.

High drywell temperatures and resultant penetration seal degradation can be prevented by drywell spray. The potential for molten core debris to spread across the drywell floor and fail the diaphragm between drywell and wetwell, or the containment liner, may also be reduced by spray operation. Drywell spray will also contribute to decontamination of the drywell atmosphere even for sequences with substantial suppression pool bypass.

The atmosphere in a Mark II primary containment is continuously inerted (by introducing nitrogen and thereby lowering the oxygen concentration) during operation, which prevents hydrogen combustion. This is a very significant mitigative feature, which is important to maintain during a severe accident. For example, wetwell venting and drywell spray operation could result in a vacuum in the containment, which could introduce additional oxygen and thus deinert the containment atmosphere.

An area of significant phenomenological uncertainty (refer to Section A.2) relates to core meltdown with the primary system at high pressure. If molten core materials are ejected from the RPV under pressure, it has been suggested in the Sandia National Laboratories (SNL) analyses that the materials form fine aerosols, which could be dispersed into the containment atmosphere and directly heat it. This could result in a large pressure pulse, which could threaten containment integrity at the time of RPV failure. BWRs have an automatic depressurization system (ADS) which can prevent high-pressure core-meltdown (on RPV low water level after a time delay or manually by the operator). The ADS has a dual role as a core-melt prevention system as well. For those sequences in which the high-pressure injection systems fail (TQU), the ADS can depressurize the reactor vessel and allow the low-pressure systems to inject water into the RPV.

Finally, the BWR Mark II primary containment is completely enclosed in a reactor building. This building is, therefore, available as a secondary containment to trap any fission products that might be released from the primary containment during a severe accident. The amount of fission products that might be trapped in the reactor building is uncertain, but both IDCOR and

SARRP analyses for the Peach Bottom reactor building indicated that it is a potentially important mitigating feature.

3.1.3 Maintain Containment Integrity and Suppression Pool Effectiveness

The above discussion has identified several plant features of a BWR with a Mark II containment that have the potential to help achieve Goal 1, namely, mitigating fission-product releases. From the above discussion, two guidelines have been developed and related to these features that will aid in assessing whether specific plants meet Goal 1. The guidelines address containment integrity, the effectiveness of the suppression pool, and the various mechanisms for possible pool bypass. As long as the dominant release path is through the suppression pool, the consequences of core-melt accidents were shown (refer to Table A.19) to be reduced by at least a factor of 10 relative to sequences that bypassed the pool. Guideline 2 also addresses one of the dominant pool bypass sequences (e.g., melt-through of the drywell to wetwell barrier) to ensure that the release is substantially reduced or that the frequency is kept low.

The Limerick Generating Station (LGS)[1] and Shoreham Nuclear Power Station (SNPS)[2] PRAs and the BNL reviews[3,4] of these PRAs confirm the importance of station blackout (SBO) sequences to the core-melt frequency. The SNPS PRA also indicates that anticipated transients without scram (ATWS) is an important contributor to core-damage frequency. The LGS PRA indicates a very small contribution from ATWS because of several ATWS related enhancements (alternate rod insertion and automatic, two train stand-by liquid control) but this does not appear typical of most Mark II plants. Thus, it is believed that the effectiveness of the suppression pool must be maintained for both ATWS and SBO.

For sequences that threaten the containment by overpressure, wetwell venting has the potential to preserve the containment function by relieving noncondensable gases and/or saturated steam, thus preventing further pressure buildup while forcing fission products to be scrubbed by the pool. However, for some dominant sequences (SBO and ATWS), existing venting procedures will be difficult to perform. For SBO sequences, power dependencies may preclude actuation of venting from the control room, and high radiation levels may hamper local manual valve actuation. For ATWS sequences, the large venting capacity requirements, short time frame for operator action and possible problems with normal isolation systems make successful venting under such conditions operationally difficult. Detailed criteria are developed for this guideline in Section 4.1.1 to help evaluate venting capability for ATWS sequences.

The SARRP event trees for Peach Bottom (refer to Volume 1) indicated that core debris melting through the steel containment shell was a dominant suppression pool bypass mechanism. As discussed above, while containment shell penetration seems unlikely for Mark II's, some present Mark II pedestal designs would allow core debris to remain out of reach of drywell sprays and eventually open paths to the wetwell airspace. Therefore it appears prudent to extend spray capability to the pedestal region and fill some vulnerable portions of pedestal cavities with concrete and steel. Other possible locations where core debris may contact and fail the drywell to wetwell barrier are the steel downcomer pipes and the outer diaphragm seal. If both

overpressure failure and drywell to wetwell melt-through can be prevented, the likelihood of pool bypass can be reduced substantially.

For sequences that still result in suppression pool bypass, the containment sprays will tend to wash out aerosols from the containment atmosphere and thus reduce the airborne fission product concentration during core-concrete interaction. In some sequences such as SBO, drywell and wetwell sprays would not be available because of ac power requirements. For some other dominant sequence class (ATWS), these sprays may not be available because of suppression pool heatup and its effects on the net positive suction head (NPSH) of the spray pumps. The guidelines and criteria developed in Section 4.1.1 address alternative power supplies and suction sources to ensure that drywell sprays will be available for the two dominant sequences.

3.2 Control the Frequency of High-Consequence Sequences

The plant features identified in Section 3.1 can effectively mitigate fission-product releases for the broad classes of accident sequences that were found to be important for the core-damage frequency (CDF). However, two accident sequences were identified in the Mark I study (refer to Volume 1) which also apply to BWRs with Mark II containments and for which substantial reducing fission-product release cannot be ensured.

The first accident sequence that may defeat the plant containment features identified in Section 3.1 is the interfacing LOCA sequence. Although none of the Mark II PRAs reviewed in Appendix A indicate that it is a significant contributor to core-melt frequency, the BNL review[4] of the SNPS PRA has identified it as a significant contributor to risk. It has also been identified as a generic issue (GI-105) by the NRC. Thus, Guideline 3 and associated criteria are developed in Section 4.2.1 to ensure that other Mark II plants review the potential contribution of interfacing systems LOCA to risk. This guideline and criteria should be considered appropriate pending resolution of GI-105.

The second accident sequence that may defeat several of the plant features identified in Section 3.1 is an ATWS with a power transient. In this sequence, the operator fails to control the RPV injection at low pressure during an ATWS event. The rising water level in the reactor vessel produces a power transient that cannot be controlled by the normal containment heat removal systems. The containment will pressurize rapidly and may fail with the resultant loss of coolant injection and eventual core melt into a failed containment. The ability to control this rapidly progressing sequence by wetwell venting is difficult, and thus mitigating this sequence appears to be unlikely. Therefore, the risk of this subset of the ATWS sequences must be controlled by ensuring that its frequency is low. In Section 4.2.2, a guideline and criteria are developed related to operator actions during an ATWS event.

3.3 Reduce High Core-Damage Frequency Sequences

In Section A.1 it was found that only a few accident sequences figure prominently in the core-damage profiles of the LGS PRA[1] and the BNL review[3] of the LGS PRA. In contrast the core-damage profile developed from the SNPS PRA,[2] the BNL review[4] of the SNPS PRA and the SNPS IDCOR-IPE[5] methodology application revealed that the profile is made up of a large number of sequences,

each making a small contribution. However, grouping the accident sequences into various sequence types enables one to determine which types are the dominant contributors to the overall CDF. Therefore, if the frequencies of the relatively small subset of dominant accident sequences in the LGS profile or the safety function failures characterized by the few dominant sequence types of the SNPS profile can be controlled, then the overall CDF can also be controlled in these and other Mark II plants.

For the LGS PRA, the dominant sequences are the TQUX sequence and the station blackout sequence, TB, (specifically, $T_E UV$). Contributions to the total CDF by the TW sequence (failure of containment heat removal) and the TC sequence (ATWS) are relatively minor, partly because of improved plant design and the implementation of two accident mitigating systems (alternative rod insertion (ARI) and alternate-3A) assumed in the LGS PRA.

The TQUX sequence and the TB (station blackout) sequence are also important in all three SNPS PRA studies outlined in Section A.1. In contrast to the LGS studies, however, the ATWS sequence and the TW sequence were identified as significant contributors to the total CDF by both the SNPS PRA and the BNL review of the SNPS PRA. These SNPS PRA studies also found potential significant contribution from accident sequences initiated by internal flooding, failure of water-level instrumentation, and the loss of reactor building service water. Some of these accident sequences, however, are plant specific to Shoreham, and were not analyzed in the LGS PRA or the BNL review of the LGS PRA.

3.3.1 Reactor Pressure Vessel Depressurization Performance

As mentioned in Section 3.1.2, the ADS is an important system for mitigating loss of high-pressure injection sequences. For both LGS and SNPS, failure to actuate the ADS in a timely manner was determined to be one of the most important contributors to total CDF. According to an estimate made by the NRC staff (NUREG-1068),[6] implementing an improved ADS initiation logic alone can reduce the CDF for the LGS by at least a factor of two. Guidelines and criteria are provided in Section 4.3 which will help ensure that other Mark II plants have a low TQUX frequency. Additionally, the ability to depressurize the RPV is an important mitigative feature that helps maintain suppression pool effectiveness (Guideline 7).

3.3.2 Station Blackout

In both the LGS PRA and the BNL review of the LGS PRA, station blackout (SBO) was identified as one of the primary contributors (40% and 32% respectively for the LGS PRA and the BNL review) to total CDF. As summarized in Table A.4 of Section A.1, loss of off-site power occupies a dominant position among the various sequence initiators contributing to the total CDF in all three SNPS PRA studies. A thorough examination of the relevant event trees further reveals that large fractions (72% for the SNPS PRA, 90% for the BNL review, and 66% for the Shoreham IDCOR-IPE methodology application) of the CDF contributions listed under loss of off-site power in Table A.4 can be attributed to SBO. Therefore, a severe accident guideline with specific criteria has been developed in Section 4 related to these accident sequences. SBO is currently the subject of an unresolved safety issue (USI A-44).

Station blackout (SBO) refers to a loss of the offsite power supply with concurrent failure of the two emergency ac power divisions. Reducing SBO sequences is addressed by the proposed NRC SBO rule. The guideline and associated criteria developed by the present study emphasize the need to search for plant specific features and potential common cause failures which could disable systems required to work during an SBO. For individual plants which are found to have a vulnerability to SBO, the criteria highlight the importance of proper emergency procedures and operator training in recovering from an SBO event.

For the BWR-4/5 design, the two systems designed to operate in the presence of a station blackout are the high-pressure coolant injection (HPCI in BWR-4's) or high-pressure core spray (HPCS in BWR-5's) and the reactor core isolation cooling (RCIC) systems. By removing the long-term SBO sequence related to ac dependent failure modes of either system, the SBO CDF can be significantly reduced.

3.3.3 Loss of Containment Heat Removal

Accident sequences involving loss of containment heat removal (e.g., TW) were found to be quite important in the earlier PRA studies such as the Reactor Safety Study (RSS)[7] and the Browns Ferry IREP.[8] In those studies, the TW sequences accounted for 53% and 50% respectively of the total CDF. Owing partly to differences in the PRA methodology employed, as well as to differences in the plant design and operation of safety systems, the TW sequences were determined to be less important in the LGS PRA (about 6%) CDF. For the SNPS PRA, however, the TW sequences still contribute significantly (about 15.5%) to the CDF. Therefore guidelines have been developed in Section 4 to control the frequency of TW sequences.

There are a number of factors contributing to the reduction in the relative importance of the TW sequences for the Limerick station. The containment heat removal system in Limerick is more reliable than that of the BWR plant considered by the RSS. Furthermore, the methodology employed in the LGS PRA is more detailed and realistic, particularly because it takes into account the possibility of recovering from a failed state in the containment heat removal (CHR) system. Also, when the RSS was performed, it was assumed that overheating of the suppression pool failed emergency core cooling (ECC) injection and therefore the containment failed with a conditional probability of unity. ECC injection failure came about either by failure to maintain NPSH conditions for the ECC pumps because of the heated pool or, surviving that, loss of their suction source by some overpressure failure of the containment itself. Since that early study, investigations into ECC pump survivability have demonstrated that the pumps have a substantial likelihood of successful operation given a heated suppression pool. In particular, the emergency pumps in the Limerick station have improved NPSH capability.

The concern over containment failure can further be mitigated by installing a containment (wetwell) venting system (refer to Table 4.1). As discussed in the BNL review of the LGS PRA, the availability of such a system can exert considerable influence on reducing the CDFs associated with the TW sequences. Establishing alternate sources of injection capability can also avoid reliance upon an overheated suppression pool (refer to Table 4.6).

3.3.4 Support System Interdependencies

Most PRAs have stressed the importance of unrecognized interdependencies having the potential to compromise the performance of many critical safety systems. In many cases, risk assessment studies have identified such vulnerabilities very early in the study and "fixes" have been made which substantially reduce risk. The vulnerability of ECC equipment to flooding has been identified for the SNPS and "engineering judgement" indicates that it may be useful to search for the existance of such interdependencies in other Mark II plants.

3.3.5 Flooding Within the Reactor Building

One of the accident sequences, whose potential for contributing to the CDF was specifically evaluated in the SNPS PRA, is the release of excessive water into the reactor building (RB). Both the SNPS PRA and the BNL review of the SNPS PRA revealed that accident sequences induced by such an initiator contribute substantially to the CDF (3.9×10^{-6} and 2.0×10^{-5}, respectively).

At the SNPS, the majority of safety-related equipment is located throughout the RB, which surrounds the Mark II containment structure. The largest concentration of safety-related equipment, including all of the emergency core cooling system (ECCS) pumps, is located in the elevation 8 compartment (the lowest level of the RB). Although such an arrangement has the advantage of easy maintenance, good personnel access and the capability for compartment ventilation by natural circulation, there is also a remote possibility of a common-mode event disabling all the equipment in the compartment. Large water leakage from equipment in the reactor building, for example, will drain to the elevation 8 compartment via openings or stairwells, causing a high water level which may disable the ECCS equipment.

In view of the potentially significant impact, the SNPS PRA's evaluation of the CDF because of RB flooding was accorded a special review by BNL. Three flooding depths (1 ft. 3 in., 1 ft. 10 in., and 3 ft. 10 in.) were determined to be crucial to the unavailability of various ECCS equipment. The initiator event trees were, therefore, revised accordingly.

To help ensure that other Mark II plants with a similar safety-related equipment layout have low sequence frequencies because of RB flooding, guidelines and criteria are provided in Section 4.3. In particular, two issues identified by the BNL study are addressed: The random failure of an equipment protection circuit breaker and the failure of Shoreham Plant Procedure Guides to require a systematic check of system parameter indicators in the control room following a RB flooding alarm annunciation.

3.4 References for Section 3

1. "Probabilistic Risk Assessment - Limerick Generating Station," Philadelphia Electric Co., 1982.

2. "Probabilistic Risk Assessment - Shoreham Nuclear Power Station," Long Island Lighting Company, June 1983.

3. I. A. Papazoglou et al., "A Review of the Limerick Generating Station Probabilistic Risk Assessment," Brookhaven National Laboratory, NUREG/CR-3028, February 1983.

4. D. Ilberg et al., "A Review of the Shoreham Nuclear Power Station Probabilistic Risk Assessment (Internal Events and Core Damage Frequency)," Brookhaven National Laboratory, NUREG/CR-4050, June 1985.

5. A. D. Bunch et al., "IDCOR Individual Plant Evaluation Method Applied to the Shoreham Nuclear Power Station," Long Island Lighting Company, April 1986.

6. "Review Insights on the Probabilistic Risk Assessment for the Limerick Generating Station," U.S. Nuclear Regulatory Commission, NUREG-1068, August 1984.

7. "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, NUREG-75/014, October 1975.

8. S. E. Mays et al., "Interim Reliability Evaluation Program: Analysis of the Browns Ferry, Unit 1, Nuclear Plant," Idaho National Engineering Laboratory, NUREG/CR-2802, July 1982.

4. GUIDELINES AND CRITERIA FOR A BWR WITH A MARK II CONTAINMENT

In Section 3, those accident sequences and sequence-types that dominate the core-damage frequency (CDF) were identified as were those that are potentially of high consequence. Vulnerabilities of the Mark II containment to severe accident containment loads were discussed and those features of a BWR with a Mark II containment (BWR Mark II), which are important for preventing core damage and available for mitigating fission-product releases to the environment were identified.

Based on the "insights" from previous PRA studies and other severe accident sequences, the following sections provide guidelines defining "deterministic, plant-specific guidance on the design features and operating characteristics which are to be examined by the utilities,"[1] and criteria defining "deterministic standards for judging the acceptability of plant features."[1] From SECY-86-76,[2] further guidance is provided in defining guidelines and criteria. These guidelines "will specify the plant features and operator actions which are considered important to ensuring acceptable risk for the reference plant."[2] Further acceptance criteria (for the various guidelines) "will specify the attributes necessary to ensure acceptable performance."[2]

Based on this work, nine guidelines were developed which reflect the importance of these features to plant risk. As discussed in Section 2.2.1 these guidelines indicate areas of potential improvements for various areas of plant design and operation of which utilities should be aware when conducting assessments. It is further noted that a number of the guidelines appear to overlap various generic issues as defined by the NRC. Final resolution and disposition of these generic issues may encompass NRC-imposed requirements. However, the guidelines and criteria presented herein are intended only for the purposes noted above. The guidelines are summarized in Table 1.1.

Guidelines 1 and 2 were developed to ensure the capability to mitigate fission-product releases (Goal 1) with reference to maintaining containment integrity and maintaining suppression pool effectiveness.

Guidelines 3 and 4 were developed for controlling the frequency of high-consequence sequences (Goal 2) with reference to minimizing interfacing systems LOCA frequency and mitigating anticipated transients without scram (ATWS) sequences.

Finally, Guidelines 5 through 9 were developed for reducing high core-damage frequency sequences (Goal 3) with reference to mitigating station blackout (SBO) sequences, mitigating loss of containment heat removal sequences, enhancing reactor pressure vessel (RPV) depressurization performance, examining support system interdependencies, and mitigating floods within the reactor building.

The remainder of this section is organized into three subsections corresponding to the three basic goals. In each subsection, the corresponding guidelines are discussed from which detailed criteria are developed in order to provide standards by which each plant could be measured for compliance with the guidelines. The criteria address (see Section 2.2.2), under severe accident conditions, the general issues of (1) survivability of equipment (i.e., whenever credit is given for a system or a component to mitigate the accident,

the ability of the equipment to function under the environmental conditions and fluid dynamic loads associated with severe accident sequences must be taken into account), (2) equipment capabilities, capacities, and duration of operability, (3) accessibility of equipment, (4) availability of support systems, (5) identification of necessary components, (6) identification of important operator actions, and (7) identification of parameters for initiation of mitigating systems and operator actions.

4.1 Mitigate Fission-Product Releases

In order to minimize off-site consequences for the dominant core damage sequences, the BWR Mark II containment systems (both primary and secondary) should be able to retain a substantial fraction of fission products released even under severe accident conditions.

4.1.1 Maintain Containment Integrity and Suppression Pool Effectiveness (Guidelines 1 and 2)

As discussed in Section 3, the most important systems for mitigating high-consequence sequences are the containment and its suppression pool. In addition to condensing the steam generated in an accident, the suppression pool also acts to remove fission products from the containment atmosphere. As long as any release path is forced through the pool (e.g., during wetwell venting), the pool will act to reduce the environmental release fractions by a factor of 10 or more. Thus, these mitigative guidelines deal with maintaining containment integrity, ensuring the effectiveness of the pool as a fission product mitigation system.

Tables 4.1 and 4.2 provide criteria which may be used to evaluate each plant's capability to avoid breach of the containment and suppression pool bypass and possible suppression pool bypass mechanisms that were identified in Section 3.

4.2 Control the Frequency of High-Consequence Sequences

In Section 4.1, guidelines and criteria were developed that should effectively ensure containment integrity and mitigate fission-product releases for the broad classes of accident sequences that were found in Appendix A to be important to the core-damage frequency. However, two accident sequences were identified for which the BWR Mark II containment has limited means of mitigating fission-product releases; namely an interfacing systems LOCA and an ATWS with a power transient. In this section, guidelines and criteria for controlling the frequency of occurrence of these potentially high-consequence sequences are developed.

4.2.1 Interfacing Systems LOCA (Guideline 3)

In general, BWR PRAs have found the interfacing systems LOCA (ISL) to be a highly unlikely event (less than 10^{-7}/reactor year). However there are some BWRs (e.g., Shoreham) for which the ISL is risk significant because of the potentially high releases. The objective of this guideline and associated criteria is to ensure that the frequency of ISL events is kept at an acceptably low level. BNL is presently performing a study to provide technical support to the NRC for the meaningful resolution of the generic issue related

to ISL (GI-105). Therefore, the criteria for this guideline should be considered appropriate pending resolution of the generic issue.

In order to control the frequency of ISL sequences, specific performance criteria have been developed to assess the performance of equipment, systems and operators. The criteria relate to equipment (low-pressure systems interfacing with high-pressure systems) and operator performance (isolation and relief valve maintenance and surveillance).

Detailed criteria developed for this guideline are given in Table 4.3.

4.2.2 Anticipated Transients Without Scram (Guideline 4)

The important attributes of the ATWS sequence with respect to operator actions were found[3] to be the likelihood of misleading instrumentation, the need to inhibit automatic safety systems, the use of required mitigating actions which conflict with operator response to other accident conditions, and the need for coordinated actions and communications among control room crew members under highly stressful conditions.

For the ATWS guideline, the performance of equipment, systems, and operators should be assessed against specific performance criteria to ensure successful use of this guideline. The criteria relate to the equipment, systems, and operator performance by emphasizing operator familiarization, aids, and understanding of potentially conflicting signals.

Detailed criteria developed for this guideline are given in Table 4.4 and are based upon the assumption that each of the plants is (or will be) in compliance with the NRC rule on "Reduction of Risk from Anticipated Transients Without Scram for Light-Water-Cooled Nuclear Power Plants."[4]

4.3 Reduce High Core-Damage Frequency Sequences

The major contributors to the core-damage frequency (CDF) for both the LGS and SNPS were presented in Section 3.3. The PRA analyses performed for these plants indicate that the TQUX sequences (failure to depressurize the RPV for injection with low pressure systems) and the SBO sequences are prominent contributors to the CDF. The results of the SNPS PRA and the BNL review of the SNPS PRA further suggest that besides these two types of sequences, other sequences, namely, the ATWS sequences and the loss of containment heat removal (CHR) sequences (i.e., TW sequences) can also be the major contributors to the CDF. The potentially significant contribution to the CDF from sequences initiated by release of excessive water into the reactor building was also identified.

4.3.1 Station Blackout (Guideline 5)

In most PRAs for light-water-reactors (LWRs), station blackout (SBO) sequences have been major or prominent contributors to the CDF. As part of the effort to resolve the unresolved safety issue (USI A-44), the NRC is proposing to amend its regulations "to provide further assurance that an SBO (loss of both offsite power and onsite emergency ac power systems) will not adversely affect the public health and safety."[5] For accident sequences, developed by an individual plant examination (IPE), which involve the loss of offsite power

and onsite emergency power, the proposed SBO rule should be examined for applicability. The criteria associated with Guideline 5 are intended to emphasize the need to search for plant specific features and potential common cause failures which could disable systems required to work during an SBO. For individual plants which are found to have a vulnerability to SBO, the criteria given in Table 4.5 highlight the importance of proper emergency procedures and operator training in recovering from an SBO event.

4.3.2 Loss of Containment Heat Removal (Guideline 6)

For some of the PRAs and the PRA reviews examined in this study, sequences with successful coolant injection but with subsequent loss of containment heat removal (CHR) (i.e., TW sequences) can be important contributors to the CDF; in those PRAs it is assumed that containment failure causes loss of ECC injection. As discussed in Section 3.3.3, in the PRAs wehre those sequences are not important, the main factor for the low contribution to CDF is because of credit given for containment venting and alternative sources of injection. Therefore it appears to be important to have alternate injection sources available in addition to wetwell venting to provide adequate CHR during accident sequences with successful ECC injection but with subsequent loss of CHR.

Detailed criteria developed for this guideline are given in Table 4.6.

4.3.3 Reactor Pressure Vessel Depressurization Performance (Guideline 7)

In the LGS PRA and the BNL review of the LGS PRA, sequences with failure to depressurize the reactor pressure vessel (RPV) after failure of the high-pressure injection systems (TQUX sequences) are important contributors to the CDF. A similar trend was also found in the SNPS IDCOR-IPE methodology application and, less obviously, in the SNPS PRA and the BNL review of the SNPS PRA. In the LGS PRA, the automatic actuation of the ADS only occurs on coincident signals of "high" drywell pressure and "low" reactor vessel water level. For a large number of transients with loss of high-pressure injection, these coincident signals will not occur. Therefore, the contribution of these sequences to CDF is dependent upon the intervention by the operator to manually depressurize the reactor. The failure probability for the safety function X (e.g., timely actuation of ADS) was assigned a value of 2.0×10^{-3}/demand and 6.0×10^{-3}/demand respectively in the LGS PRA and the BNL review of the LGS PRA. As pointed out in Section A.1, the CDF can be substantially reduced by changing the ADS auto-actuation logic to eliminate the need for the "high" drywell pressure signal. SNPS has already adopted this improved logic so that the ADS will be automatically initiated if "low water" level (Level 1) occurs. To reflect this change, a smaller failure probability for X (8.4×10^{-4}/demand) was employed in the SNPS PRA. The predominance of the TQUX sequences in the core-damage profile of SNPS IDCOR-IPE methodology application is partly because of the higher failure probability (2.4×10^{-3}/demand) assigned to X despite the logic improvement.

Detailed criteria developed for this guideline are given in Table 4.7.

4.3.4 Support System Interdependencies (Guideline 8)

One of the primary benefits of performing a rigorous PRA is that the system interdependencies are modeled and are reflected in the results. However, not all PRA studies have performed rigorous interdependence analyses and therefore have not ferreted out all of the possible subtle interdependencies. This may have profound effects upon their results. A dependency is defined as the failure of one system leading directly or indirectly to the failure of another system.

An in-depth application of basic PRA methodology with respect to interdependencies yielded significant findings on a previously heavily studied PWR. To illustrate this point, reference is made to the BNL study of system interactions (support system interdependencies) at Indian Point Unit 3.[6] The major finding of that study was that a specific single station emergency battery could fail and among other things, negate the entire low-pressure injection function. The point to be emphasized here is that none of the numerous other studies and reviews of the Indian Point 3 design were able to detect this important single failure nor did the BNL study until all the support systems were explicitly modeled, linked together (the fault tree linking approach[7]) and solved using the SETS computer code.[8]

NUREG-1150[9] has provided a thorough application of the latest PRA methods to five reference plants and the results point out numerous insights into the importance of specific design differences among the studied plants. However, the NUREG-1150 authors emphasize the importance of support system differences and the difficulty of extrapolating the result from one plant to another.

It is not sufficient to make a single overall dependency table of the front-line and support systems for a given plant and simply compare that to the reference plant. No two plants will have the same set of system interdependencies. Support systems vary widely from plant to plant even though the plants may be of a similar class and have the same set of front-line systems.

It is recognized that following the steps outlined in Table 4.8, in a rigorous fashion, is a major undertaking. This fact, however, does not diminish its importance.

Based upon the dominance of the SBO sequences to the BWR designs, it is also recommended that a detailed interdependency table be constructed for this sequence with all dependencies conditioned upon the existence of an SBO for various lengths of time. This table should also explicitly identify all of the expected failure mechanics (e.g., identify whether battery failure is because of loss of room cooling or charge depletion).

4.3.5 Flooding Within the Reactor Building (Guideline 9)

Although medium or small leakages can be adequately mitigated by the existing sumps or pumpback systems, large water leakages are of primary concern in reactor building (RB) flooding. Potential water sources for excessive water release into the lowest RB level include the suppression pool, the condensate storage tank, the reactor coolant system, the service water system and the fire protection system storage tank. Some of the major equipment located in the lowest RB level compartment may include emergency core cooling (ECC)

pumps and their electrical control panels for high-pressure coolant injection (HPCI), RCIC, core spray and low-pressure coolant injection (LPCI).

RB flooding can be initiated by (1) a major maintenance which requires exposing a safety system to the RB atmosphere, and (2) breaks in the pressurized or the non-pressurized part of piping or components. In item 1, "major maintenance" refers to those actions which would require dismantling of system components thus eliminating a barrier between large sources of water and the RB. RB flooding can partly be prevented and/or mitigated through proper training and procedures. For example, once the RB is flooded, the operator should be able to follow the instructions for responding to the alarm to identify the source of the flood and isolate it before the water level in the lowest compartment reaches the critical level. The operator should also know about alternative devices or equipment which can be utilized to provide coolant injection to the reactor vessel in case of emergency core cooling (ECCS) systems equipment failures in the flooded compartment.

The BNL study[10] of SNPS revealed that although the SNPS Alarm Response Procedures give general guidelines for monitoring system parameters to determine the leakage location and initiate the leakage isolation, specific requirements for operators to systematically check the operation parameters of relevant systems are not included. BNL also identified that the random failure of an equipment protection electric circuit breaker coinciding with RB flooding may result in the propagation of failures to the upstream motor control center (MCC), other MCCs, and the associated load centers. It is important that this potential common-mode failure be avoided.

Detailed criteria developed for this guideline are given in Table 4.9.

4.4 Using the Guidelines and Criteria

Numerous investigations, including PRAs, have been performed for the reference plants and for similar plants by both the NRC and the nuclear power industry. The insights gained from many of the studies have been used in developing the guidelines and criteria contained in this report (including the other volumes relating to other plant types). The guidelines and criteria are issued to guide the analyst performing an IPE. This guidance is in the form of plant features, operator actions and the criteria for assessing those features and actions found to be helpful in reducing the overall risk for Limerick and Shoreham and other Mark II plants. Thus, the guidance is given to provide a resource in examining the subject plant to determine if the same, or similar, guidelines will be of value in reducing overall plant risk. These guidelines and criteria are intended to be used solely as guidance, but they may include (as a subset) some requirements generated by the NRC on generic issues.

4.5 References for Section 4

1. R. Barrett, "Status of the Severe Accident Program for Operating Reactors," NRR Staff Presentation to the ACRS Subcommittee Class 9 Accidents, February 24, 1986.

2. SECY-86-76, "Implementation Plant for the Severe Accident Policy Statement and the Regulatory Use of the New Source-Term Information," NRC/EDO, February 28, 1986.

3. W. Luckas et al., "A Human Reliability Analysis for the ATWS Accident Sequence With MSIV Closure at the Peach Bottom Atomic Power Station, Brookhaven National Laboratory, Technical Report A-3272, April 1986.

4. ATWS Final Rule - Code of Federal Regulations, Title 10, Section 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram Events for Light-Water Cooled Nuclear Power Plants," June 1984.

5. NRC Station Blackout Proposed Rule, Federal Register, Volume 51, No. 55/March 21, 1986, pgs. 9829-9835.

6. R. Youngblood et al., "Fault Tree Application to the Study of Systems Interactions at Indian Point 3," Brookhaven National Laboratory, NUREG/CR-4207, January 1986.

7. American Nuclear Society and Institute of Electrical and Electronics Engineers, "A PRA Procedures Guides," NUREG/CR-2300, January 1983.

8. R. B. Worrell and D. W. Stack, "A SETS User's Manual for the Fault Tree Analyst," Sandia National Laboratories, NUREG/CR-0465, SAND77-2051, November 1978.

9. "Reactor Risk Reference Document," U.S. Nuclear Regulatory Commision, NUREG-1150, Draft for Comment, February 1987.

10. K. Shiu et al., "A Review of the Accident Sequences Following an Excessive Release of Water at Elevation 8' of the Reactor Building in the Shoreham Nuclear Power Station," Draft Report, NUREG/CR-4049, BNL-NUREG-51835, March 1985.

Table 4.1 Criteria for BWR Mark II Containment
Guideline 1: Maintain Containtment
Integrity

Concern: Breach of the containment boundary in the progression of a severe accident can lead to significant releases of radioactivity.

Functions: Wetwell Venting of Noncondensable Gases - (Guideline 1.A)

Guideline 1.A. Provide Wetwell Venting

Basis: Implementation of wetwell venting will significantly reduce the potential for loss of containment integrity because of overpressurization events.

Caution: Containment venting should not be indiscriminately performed. A clear understanding of the accident sequence in progress should have been attained before initiating venting. The effects of venting should have been assessed and made known to the operators during the training program. The assessment should include the effects of containment venting on the operation of ECC injection systems and health consequences.

Criteria:

1.A.1. For accident sequences where wetwell venting has been assessed to be beneficial, wetwell venting should commence, except for a station blackout, when containment pressure reaches the predetermined containment venting pressure setpoint. In selecting the containment venting pressure setpoint, the following functions should be ensured:

a. The ultimate containment pressure capability would not be exceeded.

b. The backpressure acting on the safety relief valve assemblies would not prevent them from performing their function.

c. The vent valve assemblies would not be prevented from performing their function.

During a station blackout, wetwell venting should commence in accordance with the criteria developed using the BWR Emergency Procedure Guidelines (EPG), i.e., following the onset of the transient and before depletion of the station batteries. If station batteries are not available, the capability of manual initiation of wetwell venting should be assessed (see Criterion 1.A.2).

1.A.2. If manual initiation of wetwell venting is deemed necessary, the time required to perform this function should be taken into account in the training and procedures to preclude the potential for exposing personnel to harsh environment. Otherwise, the containment venting valve(s) should be capable of being remotely actuated during a station blackout.

1.A.3. Operator training and emergency procedures should specify the plant parameters that will prompt the operators to make preparation, commence and terminate the venting sequence. The training and procedures should also be consistent with the required actions and timing of those actions so venting will commence immediately when required (see Criteria 1.A.1 and 1.A.2). The training and procedures should further specify the flowpath(s) available for venting, specific components to be aligned, and the required positions/states for these components. The training and procedures should specify how to proceed if it is not possible to terminate venting.

1.A.4. For each accident sequence where venting is credited (e.g., assumed to prevent containment failure) the capacity of the vent lines and associated vent valves should be assessed to determine whether the venting rate has the capability to decrease containment pressure.

1.A.5. The criteria for filtering are dependent on the potential for bypassing the suppression pool. Whether the suppression pool is bypassed or not, the radiological release should be reduced by an order of magnitude compared to no filtering. The venting flowpath should ensure that all media to be vented pass through the suppression pool thus providing filtering by the pool. If the potential of a vent path to bypass the suppression pool is high, a filter should be provided in this vent path with the ability to reduce radiological releases by an order of magnitude.

1.A.6. Equipment designated to support wetwell venting should be assessed for its ability to function reliably for a sufficient period under the predicted environmental and fluid loads associated with venting commencement pressure. If necessary, it should be enhanced to include operation during the vaporization release phase of core-concrete interaction.

1.A.7. The effects of possible hydrogen burn, radiation, and/or steam on equipment located in the reactor building outside of the primary containment should be considered in the venting assessment. If equipment important to the mitigation of accident sequences is jeopardized by venting, alternate venting paths, judged not to be detrimental, should be identified and assessed. Consideration should also be given to the effectiveness of the reactor building blowout panels and fire sprays to accommodate the discharge through the primary containment vents, thereby ensuring reactor building structural integrity.

1.A.8. The effects of possible containment depressurization on the NPSH of the ECC related pumps should be assessed. Alternate injection sources which are unaffected by venting should be considered.

1.A.9. The capability to terminate venting and the conditions under which venting would be terminated should be considered in the venting assessment. Specifically, the level of radioactivity in the wetwell airspace should be considered with regard to the projected offsite consequences.

Table 4.1 (Continued)

1.A.10.	Operator training and emergency procedures sho 1 specify the possible actions to preclude deinerting the containment by terminating venting before a negative pressure differential is reached in the vent path.

Table 4.2 Criteria for BWR Mark II Containment Guideline 2: Maintain Suppression Pool Effectiveness

Concern: Bypass of the suppression pool in the progression of a severe accident can lead to significant releases of radioactivity that would otherwise not occur if the fission products were retained in the containment and/or scrubbed by the suppression pool.

Functions: Suppression Pool Bypass - (Guideline 2.A)
Drywell Spray - (Guideline 2.B)
- Containment Heat Removal, Fission Product Scrubbing, and Debris Bed Cooling

Guideline 2.A. Prevent Suppression Pool Bypass

Basis: Implementation of the following criteria will significantly reduce the potential for bypassing the suppression pool.

Criteria:

2.A.1. The following should be considered to assess whether core debris leaving the RPV can fail the drywell to wetwell gas space barrier. Some Mark II pedestal configurations allow the accumulation of core debris in dead-end cavities, out of reach of the drywell sprays. Extensive core-concrete interaction may subsequently open a path from the drywell to the wetwell gas space. In other designs, the drywell downcomers provide a path to channel debris to the suppression pool, where it is quenched. The floor penetrations such as the downcomers, the equipment drains and the SRV lines provide potential paths to bypass the suppression pool if they fail under direct core debris attack. Downcomers are usually anchored in the drywell to wetwell diaphragm in such a way that their complete failure is less likely than the failure of equipment drains and SRV lines which are generally supported by steel flanges. Failure of such penetrations would create a direct path from the drywell to the wetwell gas space. Measures should be taken to prevent suppression pool bypass because of the failure by core debris in uncooled pedestal cavities, the failure of steel drywell floor penetrations or the failure of the seals between the floor and the wall.

Guidance: Suitable ways of avoiding the accumulation of uncooled core debris in pedestal dead-end cavities, should be considered such as to extend spray capability to such cavities or to fill them with an appropriate mass of concrete and steel. The vulnerability of drywell floor penetrations should be assessed and if necessary the penetrations should be strengthened to better resist core debris attack.

Note: It is recognized that this is an area of phenomenological uncertainty. If operation of the drywell spray can be demonstrated (refer to Section 2.B.), then the potential for molten core debris to breach the drywell to wetwell gas space barrier may be reduced and the aerosols in the drywell atmosphere will be scrubbed. However, since drywell spray operation cannot ensure protection of the drywell floor for all dominant accident sequences, alternative means of meeting this criterion should be considered.

Table 4.2 (Continued)

2.A.2. Appropriate maintenance, surveillance and emergency operating procedures and training should specify the actions to be taken (and intervals at which these actions are to be performed) to ensure that the containment isolation valves and vacuum breakers are capable of closing as required and remaining closed during severe accident conditions.

2.A.3. The projected leakage rate through the main steam isolation valves (MSIVs) should be assessed for its source term contribution in the dominant accident sequences.

2.A.4. The effect of reactor coolant system leakage accumulation and associated radioactivity in the drywell following isolation of equipment drainlines should be assessed for their source term contribution to sequences where drywell venting or failure is anticipated.

Guideline 2.B. Provide Drywell Spray

Basis: Implementation of the following criteria should aid in decontaminating the drywell atmosphere of fission products, should help control the containment pressure rise because of the decay heat load, and should promote debris cooling.

Criteria:

The following should be assessed to ensure containment heat removal capability:

2.B.1. The heat removal provided by the drywell-spray-related components should be sufficient to remove heat loads anticipated during the dominant accident sequences. These loads include but are not limited to decay heat and the chemical energy released from metalic oxidation.

2.B.2. Drywell spray should commence when the containment pressure exceeds a predetermined value calculated in accordance with the BWR EPG or before the drywell temperature reaches the value at which ADS is qualified.

2.B.3. Drywell spray should be terminated when the containment pressure decreases below a predetermined value calculated in accordance with the EPG.

Guidance: Alternate sources of drywell spray such as a diesel-driven fire pump should be assessed for their capability to provide sufficient flow and head for adequate containment heat removal.

2.B.4. The ability of equipment designated for drywell spray to function in a reliable manner under the predicted containment conditions associated with sequences for which operation of the containment spray is needed should be assessed.

Table 4.2 (Continued)

2.B.5. Operator training and emergency procedures should specify the flow paths and specific components to be aligned and their required positions for initiating drywell spray. If a backup system and/or equipment is to be utilized, operator training and procedures should identify the flowpaths and specific actions required for any temporary system cross-connections.

2.B.6. Operator training and emergency procedures should specify the plant parameters that will prompt the operators to initiate and terminate the drywell spray. Training and procedures should be consistent with the time required to align the system and components as required.

The following should be assessed in order to evaluate the capability of the drywell spray to reduce fission-product contamination of the drywell atmosphere:

2.B.7. The spray system should be assessed for its ability to cover the entire drywell volume for an adequate time, with spray droplets of an appropriate size. Such an assessment would include the total amount of water available for long-term spray operation, as well as the pressure under which the water could be supplied to the spray headers by various sources. The elevation of the spray in the drywell, the nozzle spray pattern, as well as large obstructions in the drywell below the spray headers, should be considered when assessing volume coverage.

In addition, to promote debris bed cooling the following criteria should be assessed to ensure flooding of the drywell via sprays before RPV failure:

2.B.8. The spray initiation point (Criterion 2.B.2) should be assessed to ensure that the sprays will be initiated early enough to flood the drywell floor before RPV failure for the dominant accident sequences.

2.B.9. The spray termination point (Criterion 2.B.3) should be assessed to ensure that spray termination will not allow the debris bed to reheat after it is quenched.

Table 4.3 Criteria for BWR Mark II Containment
Guideline 3: Interfacing Systems LOCA

Concern: Although the interfacing systems LOCA sequences are not considered to be leading contributors to core-damage frequency, they represent potentially high release sequences and they appear to contribute significantly to the overall risk for some Mark II plants.

Function: Maintain Reactor Coolant System Integrity

Guideline 3.A. Prevent Overpressurization of Low-Pressure Systems

Basis: Implementation of the following criteria will ensure that the frequency of an interfacing systems LOCA will remain acceptably low.

Criteria:

Note: Resolution of Generic Issue 105 (GI-105), which deals with interfacing systems LOCAs for both BWRs and PWRs, may have an impact on this guideline. Therefore, the criteria below should be considered as appropriate pending resolution of GI-105.

3.A.1. All low-pressure lines that potentially could be overpressurized should be identified and should be provided with alarms to alert the operator to the symptoms of an overpressure event.

3.A.2. The equipment designated to provide isolation and prevent overpressurization, such as the RHR line isolation valves or the low-pressure injection system check valves, should periodically undergo operability testing and local leak rate testing (LLRT).

3.A.3. The relief capability of the relief valves designated to mitigate low-pressure system overpressurization should be established. In most if not all cases, these relief valves were not sized with the possibility of an interfacing systems LOCA in mind. However, given that an interfacing systems LOCA occurs in a non-isolatable portion of a low-pressure system, there may be alternatives available to the operator such as taking advantage of additional relief valves. If such or similar actions are found to be helpful, they should be factored into the appropriate emergency procedures.

3.A.4. Operator training and procedures should specify the actions to be taken to isolate the low pressure systems identified above or to depressurize the primary system, thereby mitigating the consequences of the interfacing systems LOCA.

3.A.5. After each reactor shutdown and cooldown, the isolation function of the pressure isolation valves should be tested. These valves should not be tested under reactor operating conditions.

Table 4.4 Criteria for BWR Mark II Containment Guideline 4: Anticipated Transients Without Scram (ATWS)

Concern: ATWS sequences have been shown to be one of the leading classes of severe-accident sequences both in terms of core-damage frequency and risk.

Function: Operator Response During ATWS (Guideline 4.A)

Guideline 4.A. Provide Operator Response During ATWS

Basis: The criteria developed here are based on the assumption that each of the plants is (or will be) in compliance with the ATWS final rule dated July 26, 1984. PRA studies have shown that the predicted core-damage frequency because of ATWS is significantly lowered based upon modifications which comply with the ATWS rule. The major thrust of the ATWS rule is on the addition and/or upgrading of scram related systems and equipment to prevent an ATWS. Human reliability studies performed in support of NUREG-1150 point to potential benefits for improved operator training and procedures to mitigate the effects of an ATWS and prevent core damage from occurring. For any individual plant which may be found to be vulnerable to ATWS, the following criteria reflect added measures that emphasize the operator's role and function in mitigating an ATWS initiator.

During an ATWS sequence, the operator is required to inject boron into the reactor pressure vessel (RPV) to inhibit initiation of automatic safety systems and to attempt to manually control and mitigate the outcome of the event. In contrast, most other accident sequences are prevented or mitigated by systems which allow the operator to monitor automatic system initiation and require intervention only when a system fails to function adequately. Thus, an ATWS sequence requires operator responses that are in opposition to the highly trained responses required for the recovery and mitigation of all other off-normal and accident events. Therefore, operator training and procedures for the ATWS sequences should specifically prepare operators to perform the unique ATWS actions called for in the BWR EPG as well as in the other measures below.

Criteria:

4.A.1. Operator training and emergency procedures should specify the plant parameters that are indicative of ATWS and the actions to be taken to verify that the reactor coolant recirculating pumps have tripped automatically. Additionally, they should specify the actions to be taken if the reactor coolant recirculating pumps do not trip automatically.

Table 4.4 (Continued)

4.A.2. Operator training and emergency procedures should ensure that standby liquid control system (SLCS) injection is initiated manually, as required, during an ATWS. Operator training and procedures should also specify the plant parameters that indicate manual SLCS actuation and the actions to be taken and verification to be made to ensure that the SLCS was actuated.

4.A.3. Operator training and procedures should ensure operator familiarity with reactor water level control during ATWS.

Note: This unique control requires actions that conflict with mitigating actions for all other accidents that call for flooding the RPV to ensure the reactor core is covered.

4.A.4. Since RPV water level indicators may be inaccurate and may provide conflicting indications of the water level, operator training and procedures should provide guidance to the operator.

4.A.5. The automatic depressurization system (ADS) should be capable of being defeated by the operator during an ATWS before its automatic initiation. Operator training and procedures should address the possible reluctance of operators to defeat a safety system, in particular, the need to inhibit the ADS immediately after an SLCS initiation attempt.

4.A.6. Operator training and procedures should specify the responsibilities of operating staff crew members and clarify how information will be exchanged among them. In particular, instrumentation readings may have to be relayed between the crew member(s) operating the control boards and the senior reactor operator coordinating the crew's response to the accident.

4.A.7. The capability, of the systems and equipment required for mitigating an ATWS, to function under predicted environmental and fluid loads associated with severe-accident sequences should be assessed and to determine whether the equipment woulf be available for an appropriate time of operation.

4.A.8. An assessment should be made of the feasibility of establishing the main condenser as a heat sink by reopening the main steam isolation valves and turbine bypass valves, if possible.

Table 4.5 Criteria for BWR Mark II Containment
Guideline 5: Station Blackout

Concern:	Station blackout sequences have been shown to be one of the leading classes of severe-accident sequences both in terms of core-damage frequency and risk.
Functions:	Reactor Pressure Vessel Injection (Guideline 5.A) Containment Integrity (Guideline 1)

Guideline 5.A. Provide Reactor Pressure Vessel Injection

Basis: Significant study and research have preceded current work on severe accidents; in particular, reference is made to the rulemaking activity already under way on station blackout. It is assumed that when the station blackout rule is finalized, some requirements of the rule may be similar in form to the criteria below. Nevertheless, during an individual plant examination (IPE), it is important to highlight those areas which previous PRAs found to be important contributors to the station blackout core-damage frequency. For those specific plants which are found to be vulnerable to station blackout events, the criteria below will assist in identifying potential areas for plant improvements as well as identifying operator actions which are key to mitigating a station blackout event.

Criteria:

5.A.1. The high pressure coolant injection (HPCI) system in BWR-4's or the high pressure core spray (HPCS) system in BWR-5's and the reactor core isolation cooling (RCIC) system, are two systems intended for the purpose of RPV injection independent of ac power. However, it has been postulated that these systems cannot continue operating in the presence of a prolonged blackout. Therefore, the HPCI/HPCS and RCIC should be assessed with respect to extending their capability to function in the presence of a station blackout.

5.A.2. Operator training and procedures should specify the plant parameters indicative of HPCI/HPCS and RCIC initiation. Additionally, the training and procedures should specify the actions required to place in operation and/or ensure continued operation of these systems under station blackout conditions.

5.A.3. The capability of HPCI/HPCS and RCIC systems to function under predicted environmental and fluid loads associated with station blackout should be assessed to determine whether they are available for an appropriate time of operation.

5.A.4. Special emphasis should be placed upon the review of the ac and dc power systems to ensure that common cause failures have been eliminated to the extent practical from the design.

Table 4.6 Criteria for BWR Mark II Containment Guideline 6: Loss of Containment Heat Removal

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Concern: Failure to remove the decay heat buildup in the suppression pool (loss of containment heat removal) following a transient event has been shown to create NPSH problems for the pumps taking suction from the suppression pool and therefore, can lead to injection failure, subsequent core damage, and containment failure. The RSS indicated that this was a leading class of core-damage sequences.

Functions: Long-Term Emergency Core Cooling (ECC) (Guideline 6.A)
Maintain Containment Integrity (Guideline 1)

Guideline 6.A. Provide Long-Term Emergency Core Cooling

Basis: Implementation of the following criteria will significantly reduce the failure potential of ECC injection because of loss of the containment heat removal function (i.e., TW sequence). Maintenance of containment integrity is addressed by Guideline 1.

Criteria:

6.A.1. Operator training and procedures should specify methods and actions for heat removal via alternate injection path(s), in conjunction with wetwell venting for severe accident conditions when suppression pool temperature precludes use of primary ECC injection paths.

6.A.2. For the alternate injection path(s), it should be demonstrated that the flow would be sufficient to preclude core damage.

6.A.3. If local operation of any equipment is required, the time required to perform these functions should be consistent with the time available to help prevent core damage and account for personnel exposure to the predicted severe-accident environment.

6.A.4. The capability of equipment used for alternate injection to function under the predicted environmental and fluid loads associated with severe accident sequences should be assessed to determine whether the equipment would be available for an appropriate time of operation.

Table 4.7 Criteria for BWR Mark II Containment Guideline 7: Reactor Pressure Vessel (RPV) Depressurization Performance

Concern: Sequences in which failure to depressurize the reactor pressure vessel (RPV) after failure of the high-pressure injection systems (HPIS) have been shown to be a leading contributor to the core-damage frequency. Many of these sequences do not create the condition necessary to actuate the ADS.

Function: RPV Depressurization (Guideline 7.A)

Guideline 7.A. Provide RPV Depressurization

Basis: Implementation of the following criteria will significantly improve the response of the ADS to facilitate depressurization of the RPV in order that low-pressure systems may be used to provide ECC injection.

Criteria:

7.A.1. The RPV should be capable of being automatically depressurized as may be required for all dominant accident sequences.

Guidance: An example of an acceptable modification to BWR-4/5 designs is the design change at Shoreham wherein the high drywell pressure signal was removed from the coincidence logic for automatic ADS actuation and the time delay for actuation has been lengthened. ASEP results for Peach Bottom indicated a high likelihood of concurrent ac and dc power failure. Plants with similar vulnerabilities may consider a dedicated backup dc supply to ensure depressurization capability under station blackout conditions.

7.A.2. The ADS should be capable of initiation and operation under the environmental conditions associated with the dominant accident sequences. In particular, the ADS should be capable of operating under the maximum pressure anticipated before venting (see Criteria 1.A.1).

7.A.3. Operator training and procedures must ensure reliable manual control of RPV depressurization.

7.A.4. The capability of the systems required to provide RPV depressurization (e.g., batteries and air supplies) to function under the predicted environmental conditions and fluid loads assocated with severe accident sequences should be assessed to determine whether the equipment would be available for an appropriate time of operation.

Table 4.8 Criteria for BWR Mark II Containment
Guideline 8: Support System
Interdependencies

Concern: When conducting a PRA, IPE or similar analysis, it is imperative that the support system interdependencies be fully developed, understood and reflected in the final results. Otherwise there is no assurance that the dominant core-damage/risk sequences have been identified.

Function: Support System Interdependencies (Guideline 8.A)

Guideline 8.A. Examine Support System Interdependencies

Basis: Implementation of the following criteria will ensure that the full set of support system interdependencies have been identified and have been reflected in the results.

Note: The following criteria are easily outlined but are not easily implemented. The complex nature of a nuclear power plant makes it imperative that this area of analysis be fully examined. However, since no two plants have identical support systems this analysis can only be done on a plant-specific basis.

Criteria:

8.A.1. All systems that provide any direct support to either a frontline or support system should be identified along with its supported system.

8.A.2. Each dependency should be conditioned as appropriate as to what sequences or under what (if not all) circumstances it applies. In view of its importance, a separate station blackout dependency table should be provided which gives the available systems, their anticipated survival period and the ultimate cause (e.g., no room cooling) of their failure.

8.A.3. The dependencies should then be linked together (preferably by computer) within the analysis in order that the extent to which their influence reaches through the systems to a consequence will be discovered. This will help identify secondary dependencies to ensure that no one failure in a support system has any unknown critical outcome on other support or front line systems.

Table 4.9 Criteria for BWR Mark II Containment Guideline 9: Flooding Within the Reactor Building

Concern: An excessive water release into a portion of the reactor building (RB) outside the primary containment which houses a concentration of safety-related equipment raises the possibility of a common-mode event disabling all the equipment in the compartment. At least one plant with a Mark II containment has been identified in which the location of safety equipment, including all ECCS pumps, in the lowest RB level makes the flooding initiator a substantial contributor to CDF.

Function: Prevent or Mitigate RB Flooding (Guideline 9.A)

Guideline 9.A. Prevent or Mitigate Reactor Building Flooding

Basis: Implementation of the following criteria may reduce the potential of a common-mode failure of safety equipment because of RB flooding in Mark II containments where RB layout combines important safety equipment in low-lying portions of the RB with exposure to possible inundation.

Criteria:

9.A.1. Operator training and procedures should ensure that the operator will diagnose and isolate any flooding of the RB that occurs.

9.A.2. Operator training and procedures should ensure that the operator is aware of alternate injection sources still available if flooding causes a common-mode failure of ECCS equipment.

9.A.3. The electrical system should be assessed for the possibility of cascading failures because of flood-induced electrical shorts. Additional isolation devices (e.g., circuit breakers) should be considered, if needed.

9.A.4. Water-tight doors in the lower levels of the RB should be alarmed to notify the operator when they have been left open.

APPENDIX A

SEVERE ACCIDENT RISK INSIGHTS

A.1 Core-Damage Profiles

The principal objective of this appendix is to present a survey, within the scope of this study, of the core-damage profiles emerging from the Probabilistic Risk Assessment (PRA) of both the Limerick Generating Station (LGS)[1] and the Shoreham Nuclear Power Station (SNPS)[2] and from their PRA reviews performed by Brookhaven National Laboratory (BNL), which were reported in NUREG/CR-3028[3] (for LGS PRA) and NUREG/CR-4050[4] (for SNPS PRA). Reference is also made to the Review Insights on the Probabilistic Risk Assessment for the Limerick Generating Station (NUREG-1068)[5] published by the U.S. Nuclear Regulatory Commission. The core-damage profile developed in the recent report[6] by the Long Island Lighting Company (LILCO) entitled, "IDCOR Individual Plant Evaluation Method Applied to Shoreham Nuclear Power Station" is also briefly reviewed and cited for comparison. Discussions on the core-damage profiles for the LGS and SNPS are presented separately in Section A.1.1 and Section A.1.2. Emphasis is placed upon describing the general characteristics of the core-damage profiles and identifying the safety function failures pertaining to dominant core-damage sequences to establish the basis for the guidelines and criteria presented in this report. The major differences between the core-damage profiles obtained from the original PRAs and from the BNL reviews and the Industry Degraded Core Rulemaking Program (IDCOR) Individual Plant Examination (IPE) methodology application are described, along with the reasons behind them.

A.1.1 Core-Damage Profiles for Limerick Generating Station

A.1.1.1 LGS PRA Core-Damage Profile

In the LGS PRA study, the Reactor Safety Study (RSS) type event tree techniques were utilized to develop the potential sequences of core-damage accidents. Accident sequences were defined in terms of combinations of safety function failures, such as loss of high-pressure injection, loss of containment heat removal (CHR), or failure of reactor scram, given the occurrence of an accident initiator. For each of the accident initiators, these combinations were generated by constructing the functional event trees. The branch point split fractions of the event tree top events were calculated as probabilities of function failures with the aid of functional fault trees or function level event trees, etc. The frequency of each core-damage sequence was computed by multiplying together the failure probabilities of the functions involved in the sequence along with the frequency of the corresponding initiator.

Twelve functional event trees were constructed to model the plant responses to the various accident initiators, which were grouped into three general categories, namely, transients with successful scram, loss-of-coolant accidents (LOCAs), and anticipated transients without scram (ATWS). A total of 86 core-damage sequences were identified, 35 of which were found to have frequencies larger than 10^{-8} per reactor year. All the core-damage sequences were divided into the following classes according to the scenario and characteristic of core damage.

- Class I: Core-damage sequences characterized by loss of coolant makeup and core damage prior to containment failure.

- Class II: Core-damage sequences which involve loss-of-long-term CHR with subsequent core damage.

- Class III: ATWS sequences with rapid core damage before containment failure.

- Class IV: ATWS sequences with containment failure before core damage. Containment failure expedites the loss of coolant makeup capability, resulting in core damage.

In Table A.1, the top 20 core-damage sequences obtained in the LGS PRA study are ranked in descending order according to frequency. They account for approximately 95% of the total core-damage frequency (CDF). Several pertinent observations from this table with respect to core-damage classes and individual core-damage sequences follow:

(1) The class that contributes most significantly to core damage is the loss-of-core-coolant inventory makeup class (Class I), which contributes roughly 82% of the total CDF. Classes II and III each contributes about 6%, while Class IV contributes only 2%.

(2) When the accident sequences originating from the same initiator are grouped together, the most dominant group belonging to Class I is the sum of T_EUV and T_EUX, which contributes roughly 44% of the total CDF. These sequences are initiated by the loss of offsite power (T_E), followed by loss of high- and low-pressure injection systems (UV), or failure of high-pressure injection along with failure to perform timely manual depressurization of the reactor vessel (UX).

(3) The most dominant sequence in Class II is T_TPW, a turbine trip with failure of safety/relief valves to close (P), followed by a loss-of-long-term CHR (W).

(4) Mechanical failures of the reactor scram system following a turbine trip or loss of feedwater dominate Classes III and IV, respectively.

(5) 90% of the total CDF is attributable to the top 13 accident sequences, with 88% of these sequences categorized as Class I.

(6) Overall, the most dominant sequences are T_EUV and T_FQUX, contributing 40% and 24%, respectively, of the total CDF.

(7) The top five accident sequences, all belonging to Class I, involve failure of the high-pressure injection system (U). The most dominant sequence, T_EUV, also involves failure of low-pressure injection.

(8) Four of the top five accident sequences involve failure of timely manual actuation of the automatic depressurization system (ADS).

(9) Accident sequences initiated by loss-of-offsite power (T_E) contribute more than 45% of the total CDF.

It is worthwhile to make a few remarks on the relative importance of safety function failures displayed by these dominant core-damage sequences compared to those revealed in the RSS. In contrast to the findings made in the RSS, sequences involving transients coupled with a loss of CHR function, i.e., TW sequences, make relatively minor contributions to the total CDF. In the RSS, such accidents were found to have the highest frequency, followed in order by ATWS sequences and transient sequences involving failures of high and low pressure injection systems. The opposite trend found in the LGS PRA[1] study can be attributed to the difference in the PRA methodology employed as well as the difference in the plant design. The CHR system in the Limerick station has a higher reliability compared to that of the BWR plant considered in the RSS. After the inception of the transients, there is a period of 20 hours, during which CHR can be recovered. Unlike the RSS, the methodology employed in the LGS PRA takes into account the possibility of recovering the CHR system after initial failure. A similar explanation can be offered with regard to the relatively insignificant contribution by the ATWS sequences to the total CDF in the LGS PRA. Following the suggestion made by the NRC staff, the LGS design incorporates the alternate-3A modification in the ATWS prevention and mitigation system. Also, the modeling and quantification of the ATWS sequences in the LGS PRA attempt to be more accurate than the conservative quantification used in the RSS.

A.1.1.2 BNL Review of LGS PRA

The primary purpose of the BNL review was to examine the risk profiles presented in the LGS PRA[1] report and compare them to those identified in the RSS.[7] The review included technical assessment of the assumptions and methods used in the LGS PRA as well as reevaluation of the results and general methodological framework of the LGS PRA study. This was achieved by carrying out both qualitative and quantitative analyses of accident initiators, data bases, accident sequences leading to core damage, core-melt phenomena, fission-product behavior, and offsite consequences. The BNL reassessment of the LGS core-damage profile and revisions to the quantifications, focused upon the following areas:

(1) Addition to the system modeling of the failures of support systems (such as ac power, dc power, and service water), and estimation of the impact of support system dependencies at the accident sequence level.

(2) Modification of the various event trees and fault trees to account for common mode failures such as failures of shared hardware and process coupling between frontline systems performing the same or different safety functions.

(3) Partial modifications of the ATWS event trees and system fault trees.

(4) Reassessment of the frequencies of the transient initiators and system unavailabilities.

Within the modifications and changes mentioned above, the total CDF was fond to be 1.0×10^{-4} per reactor year, almost a factor of seven increase from the results of the LGS PRA. If the BNL reassessed frequencies of transient initiators are not used, the total CDF becomes 8.0×10^{-5} per reactor year.

This latter value is cited as the estimated LGS CDF because of internal events in the NRC staff review (NUREG-1068).

The BNL reassessment identified a total of 53 sequences, 43 of which were found to have a frequency greater than 10^{-8} per reactor year. The top 20 sequences identified in the BNL revised core-damage calculations (incorporating all the above changes) are tabulated and ranked in Table A.2, along with their ranking in the LGS PRA. As a result of including support systems dependencies, six new sequences were identified in the BNL reassessment, four of which appear in Table A.2, ($T_T(DC)$, $T_T(AC)$, $T_T(WSW)$, and $T_F(DC)$).

In the BNL revised calculations, the top six sequences were found to contribute 90% of the total CDF. In the LGS PRA, the same percentage was attributed to the top 13 sequences. The top nine sequences in the BNL review contribute 95% of the total CDF, with all but one sequence (T_FQW, ranked 7th) categorized as Class I. Class I core-damage sequences contribute 96% of the total CDF, with the remainder being made up by Class II and III sequences. Since the BNL review reclassified the sequence $T_EC_MW_{12}$ from Class III/IV to Class III there is virtually no contribution to CDF from Class IV.

The top five sequences identified in both the LGS PRA and the BNL review are identical, except for a minor reordering of the ranking. In both studies, the first two sequences T_FQUX and T_EUV, dominate the core-damage profiles, contributing 69% in the BNL review and 64% in the LGS PRA. The sequences, T_FQW (loss of feedwater with failure of long term CHR) and T_FQUV (loss of feedwater with failure of high- and low-pressure injection systems), which ranked 12th and 15th, respectively, in the LGS PRA, are ranked7th and 9th in the BNL review because of the significant dependencies existing between the initiators and the availability of the feedwater/power conversion systems (FW/PCS). The significance of the loss of support systems, such as dc electric power and service water, is reflected in sequences $T_T(DC)$ and $T_T(WSW)$, both of which are among the ten most dominant sequences in the BNL reassessment.

A concise description of the top five accident sequences, estimated to contribute 78% and 89% of the total CDF by the LGS PRA and the BNL review is given below, along with an explanation for the quantitative discrepancy between the two assessments. The frequency shown inside parenthesis corresponds to that of LGS PRA.

(1) Sequence T_FQUX, 3.7E-5* (3.63E-6)/ry.

This sequence is initiated by reactor isolation (main steam isolation valve (MSIV) closure, loss of condenser vacuum and loss of feedwater), followed by failure to restore the feedwater and condensate system, loss of high pressure injection and failure of timely manual actuation of the ADS. The larger frequency obtained by the BNL review can be ascribed to high T_FQ (loss-of-feedwater and power conversion system) dependency as well as the use of high high-pressure coolant injection (HPCI) unvailability and higher human error probability (HEP) for depressurizing the reactor (X). The higher HEP was determined prior to the the implementation of Three Mile Island (TMI) Action Plan Item II.K.3.18 regarding modifications to the actuation logic for

*3.75E-5 = 3.7×10^{-5}.

ADS and so may not be fully representative of the current plant design. The improved ADS initiation logic involves bypassing of the high drywell pressure trip and the addition of a manual inhibit switch. Implementation of this logic modification eliminates the need for manual depressurization of the reactor vessel for certain transients such as a stuck open safety relief valve or a steamline break outside containment with HPCI failure. The failure probability for depressurization of the reactor vessel, X, can thus be significantly reduced. The corresponding sequence frequency reduction is estimated to be about a factor of six for the T_FQUX sequence, according to the NRC (NUREG-1068) report.[5] The CDF can be reduced by a factor of 2.3. These comments also apply to sequence discussions 3, 4, and 5 given below, all of which involve safety function failure X.

(2) Sequence T_EUV, 3.2E-5 (5.9E-6)/yr.

This sequence, ranked first in the LGS PRA, is initiated by loss of offsite power, with concurrent loss of onsite power because of a common cause failure of all diesel generators. There is no timely recovery of ac power and the inventory makeup (high- and low-pressure injection) systems also fail. Higher initiator frequency and higher HPCI unavailability used in the BNL calculation are mainly responsible for the large increase in the frequency. As stated previously, BNL has quantified the effect of support system dependencies (ac, dc, and service water) at the accident sequence level, and this has led to a significant increase in sequence frequency. It should be mentioned that an improved design to achieve an alternate method of HPCI and reactor core isolation cooling (RCIC) room cooling during the loss of offsite power events is currently being implemented at Limerick. The NRC (NUREG-1068)[5] report estimates that it will reduce the sequence frequencies for T_EUV and T_EUX by roughly a factor of 1.2.

(3) Sequence T_EUX, 8.6E-6 (6.9E-7)/yr.

This sequence is initiated by loss-of-offsite power and involves loss-of-high-pressure injection either because of failure to recover ac power or because of random failures, followed by failure to manually actuate ADS. Higher values used for the initiator frequency, the HPCI unavailability and the HEP (to initiate the ADS) contribute to the large increase in the BNL estimated frequency.

(4) Sequence T_TQUX, 8.0E-6 (7.73E-7)/yr.

This sequence is initiated by a turbine trip, followed by loss of feedwater, failure of high-pressure injection and failure to perform timely actuation of ADS. The nearly tenfold increase in the frequency can be attributed to a higher turbine trip frequency, a larger HPCI unavailability and a higher HEP (for manual depressurization) used in the BNL reassessment.

(5) Sequence T_IUX, 4.0E-6 (6.8E-7)/yr.

The initiator of this sequence is an inadvertent opening of a relief valve. A failure of high-pressure injection and failure to manually initiate ADS follow. The sixfold increase in the frequency can be ascribed to a higher initiator frequency, higher HPCI unavailability and HEP (for manual ADS actuation) employed in the BNL calculation.

A.1.1.3 Summary of LGS Core-Damage Profile

The relative contributions to the total CDF by the dominant sequences identified in both the LGS PRA and the BNL review can be summarized as follows:

(1) The LGS core-damage profile is essentially dominated by the sequence types, TQUX and TQUV (specifically, T_FQUX, T_TQUX, T_EUX, T_IUX, T_FQUV) and the station blackout (SBO) sequence, (specifically, T_EUV). The contributions from these sequences amount to roughly 90% of the total CDF. All of them belong to Class I. The most dominant sequences are T_FQUX and T_EUV, contributing 37% and 32% (in the BNL reassessment[3]), respectively. The major safety function failures pertaining to these sequences are loss of feedwater, loss of high- and low-pressure injection systems and failure to perform timely ADS actuation. As pointed out earlier, however, if the improved ADS initiation logic and procedures to achieve an alternate method of HPCI/RCIC room cooling are implemented, the contribution to CDF by TQUX and T_EUV sequences can be significantly reduced.

(2) Contributions from sequences involving failure of the CHR (TW), and failure of reactor protection systems (ATWS) are relatively insignificant, each contributing approximately 3 to 4%.

(3) The BNL reassessment[3] identified six new sequences involving failures of support systems such as dc power, ac power, and service water. Their combined contribution to the total CDF, however, is only about 4%.

(4) Accident sequences induced by loss of offsite power contribute more than 40% of the total CDF.

A.1.2 Core-Damage Profiles for the Shoreham Nuclear Power Station

A.1.2.1 SNPS PRA Core-Damage Profile

The methodology employed in the SNPS PRA[2] for defining accident sequences and quantifying the frequency of core damage is essentially identical to that described in the previous section for the LGS PRA. The major difference from the RSS method is that, in addition to functional and systemic event trees and system fault trees, three variations of these logic trees, namely, time phased systemic event trees, functional fault trees, and functional level event trees were utilized. Twenty-one functional event trees were developed for the various accident initiators, which were divided into four general categories (transients, LOCAs, ATWS, and low frequency transients). More than 380 accident sequences were identified, 164 of which had a frequency larger than 10^{-8} (per reactor year). Of these, 71 sequences had a frequency larger than 10^{-7}, and 15 of these sequences had a frequency larger than 10^{-6}. All the core-damage sequences were grouped into the following five classes, the definitions of which differ somewhat from those described earlier for the LGS PRA.

- Class I: Core-damage sequences characterized by the loss of core inventory makeup and core damage before containment failure. Class I is further subdivided into Classes IA, IB, IC, ID, and IE. The definition of these subclasses is shown in Table A.3.

- Class II: Core-damage sequences involving loss-of-long-term CHR function, resulting in containment failure, which may be followed by core damage. Since only part of this class will result in core damage, the SNPS PRA defines this to be a core vulnerable state. In the BNL review, it is considered a core damage state, although core damage will not always occur following containment failure.

- Class III: Core-damage sequences characterized by a LOCA in the drywell. This class is further subdivided into Classes IIIA, IIIB, IIIC, and IIID, the definition of which is also given in Table A.3.

- Class IV: ATWS sequences with containment failure prior to core damage.

- Class V: Sequences involving a LOCA outside containment, bypassing the suppression pool and drywell.

The SNPS PRA estimated the total CDF to be 5.5×10^{-5} per reactor year. The respective contributions to the total frequency from the different core damage classes are summarized in Table A.4, for the various sequence initiators. (Table A.4 also shows the results of the BNL review[4] as well as results recently obtained by LILCO's applying the IDCOR-IPE methodology to the SNPS. Discussion of these latter two results and their comparison with the SNPS PRA core-damage profile will be presented in subsequent sections.)

The largest contribution to the CDF comes from Class I, loss-of-coolant makeup, with 58% of the total CDF. The top 30 dominant sequences are listed in Table A.5 in order of descending frequency. As this table shows, a striking feature of the SNPS PRA core-damage profile is the lack of dominant sequences. The profile consists of a large number of sequences, each with a small contribution to the total CDF. This is in sharp contrast with the profile of the LGS PRA where a small number of accident sequences dominate. It should also be noted that the 30 top SNPS PRA sequences only account for 73% of the total CDF.

To aid in identifying important safety function failures, the sequences listed in Table A.5 are examined according to their core-damage sequence classification rather than individually. The sequences belonging to the same class or subclass are further divided into sequence types defined by certain sequential failures of safety functions, given an initiator. CDF contributions from various sequence types are presented in Table A.6 for all the pertinent sequence classes. Though they are based only on the top 30 sequences, the relative contributions among the classes show similarity with those displayed in Table A.4, which were derived from all the accident sequences. For example, Table A.6 shows that contributions from Classes I and IV are 62% and 27%, respectively. Table A.4 shows their contributions to be 58% and 25%, respectively.

Two sequences, T_1QUX and $T_9C_MC_2$, are seen to have relatively large contributions (about 22% each) to the CDF. The former is characterized by a loss of both feedwater system and high-pressure injection functions (RCIC and HPCI) as well as failure to manually initiate timely depressurization of the reactor using ADS. The latter comprises various ATWS sequences caused by a mechanical failure of the scram system, followed by failure of adequate reactivity control through poison injection. The T_8W sequence, which involves failure of

the CHR system, and sequence types such as T_CUX, T_2DUX', and T_DDIQ, are also seen to be relatively important.

Distribution of CDF according to accident initiators is shown in Table A.7 for each of the classes. Note that the results are based on all of the accident sequences identified in the SNPS PRA.[2] For Class I, accident sequences initiated by loss of offsite power (LOOP) are the largest contributor to CDF. This is followed consecutively by sequences originated from level instrumentation, internal flooding, and loss of condenser vacuum. For Class IV (ATWS sequences), scram failure following MSIV closure is the foremost contributor (about 55%). ATWS sequences initiated by loss of feedwater and turbine trip contribute 21% and 17%, respectively.

Table A.4 indicates that LOCA sequences are minor contributors (less than 4%) to the CDF. This is further substantiated by the fact that none of the top 30 sequences shown in Table A.5 belongs to the LOCA sequences.

A.1.2.2 BNL Review of SNPS PRA

Most of the BNL comments and modifications of the SNPS PRA pertained to accident sequence quantification. Some, however, related to the specific modeling of certain sequences. In general, BNL concurred with the overall approach taken in the SNPS PRA to develop the functional event trees. The BNL review[4] of the SNPS PRA[2] functional event trees focused on a case-by-case review of both the functional event tree accident sequence modeling and the functional event tree quantification. For the majority of the sequences, the functional event trees developed in the SNPS PRA were adopted in BNL's reassessment, with only minor modification. In the course of quantifying the accident sequences, however, some of these trees were significantly revised by BNL, including ATWS event trees and those related to the loss of reference leg, loss of service water and release of water at level 8. Details on modeling changes and the reasoning behind them are presented in Section 3.2 and the Appendices to Section 5 of the BNL review.

The quantified results of the BNL review are included in Table A.4. The total CDF estimated by the BNL review is 1.4×10^{-4} per reactor year, roughly a factor of 2.5 increase over the SNPS PRA. This increase can be attributed mainly to the increased contributions from ATWS, LOOP, transients with successful scram, and internal flooding initiators. The interfacing LOCA frequency was found to be roughly half an order of magnitude higher than the SNPS PRA estimate.

Factors contributing to the difference between the SNPS PRA and BNL CDF estimates are briefly discussed below:

(1) Some of the transient initiator frequencies in the BNL review were increased based on an updated source of experimental data. Affected by these increases were the ATWS sequences as well as the MSIV closure and turbine trip transients. For example, the contribution from transient initiators was increased by a factor of 1.7 (from 1.3E-5 to 2.2E-5) largely because of revised initiator frequencies.

(2) In the BNL review,[4] the frequency for LOOP was reevaluated to be 0.15 per year for the Shoreham site, as compared to 0.08 per year used in the

SNPS PRA.[2] This increase, however, was partly offset by the higher LOOP recovery probabilities used in the BNL calculation, which were derived from a more recent evaluation of the Licensee Event Reports (LERs).

(3) The increase in the BNL assessed LOOP initiated CDF is partly attributable to the inclusion of loss of instrumentation indications in the control room.

(4) The ATWS functional event trees were revised by taking into consideration Shoreham plant specific information. The factor of 2.5 increase in the ATWS CDF (from 1.8E-5 to 4.5E-5) can be largely ascribed to the higher initiator frequency employed in the BNL reassessment. The event tree modifications and changes in the assumptions caused only a small increase.

(5) The factor of five increase in the frequency (from 3.9E-6 to 2.0E-5) for the "excessive release of water in the reactor building" obtained by BNL is caused by (i) a higher initiator frequency used by BNL based on a more up-to-date and elaborate model, (ii) an increase in the condensate injection failure probability (0.1 instead of 0.01), and (iii) a more elaborate time-phased model for considering the early failure of HPCI and RCIC.

(6) Contributions from loss of reference leg and drywell cooling increased by roughly a factor of three (from 3.9E-6 to 1.2E-5) primarily because of two additions made by BNL: (i) the common-mode failure caused by maintenance of the second reference leg, and (ii) the miscalibration of the two sensors on the other leg. These more refined models led to the identification of several new sequences, not included in the SNPS PRA, which increased the total CDF contribution from this initiator.

(7) The CDF contribution because of a LOCA outside the drywell is about five times higher in the BNL review (2.0E-7 vs 3.7E-8), partly because of a change in the initiator frequency estimate. Although this contribution to total CDF from this sequence is very small (less than 0.2%), it may represent an important contribution to risk.

(8) The increase in the Class II contribution occurs because of additional sequences considered under loss of service water and LOOP transients. The loss of condenser vacuum also contributed to the increase.

To clarify the contribution to the total CDF from individual accident sequences, the top 31 sequences obtained in the BNL reassessment are listed in Table A.8 in order of their rankings. These sequences account for about 77% of the total CDF. It can be seen that there are no dominant sequences contributing a large fraction of the total CDF - a pattern observed in the SNPS PRA. The core-damage profile is made up of a large number of small contributors. It is worth noting that only one-third of the sequences listed in Table A.8, the BNL top 31, appear among the top 30 sequences identified by the SNPS PRA (Table 5).

The top five sequences contain three ATWS sequences identified from the BNL revision of the ATWS functional event trees. The T_TC_MKQ sequence, ranked first in the list, involves mechanical failure of the scram system following a turbine trip, failure of alternate rod insertion (ARI), and failure

of feedwater to run back in a timely manner. The other two ATWS sequences, $T_MC_MKU_H$ and $T_TC_MKU_H$, are initiated either by MSIV closure or by a turbine trip, followed by failure to scram and the operator's failure to maintain water level above level 1. The second ranked sequence, T_EIDGL, is initiated by LOOP, with no diesel generators available. Failure to recover both offsite power and diesel generators within 30 minutes, and subsequent failure in water level instrumentation lead eventually to core melt. The FS_0QUX sequence, ranked third in the BNL review (sixth in the the SNPS PRA) is the accident sequence involving a postulated release of excessive water at elevation 8 of the reactor building. Its frequency has been estimated by a separate study carried out at BNL.[7]

Contributions to CDF from sequences grouped together according to core-damage sequence types and accident initiators are presented in Tables A.9 and A.10, respectively. Note that Table A.9 is based on the 31 sequences appearing in Table A.8, whereas Table A.10 is based on all of the accident sequences identified in the BNL review.

The following observations can be made by comparing BNL sequences of Table A.9 with the SNPS PRA sequences of Table A.6:

(1) The T_1QUX sequence is the leading contributor to CDF in the BNL review. T_1QUX and the ATWS sequence, $T_9C_MC_2$, are most important in the SNPS PRA, contributing equally to CDF.

(2) Contributions from the three sequence types, $T_TC_MKU_H$, T_TC_MKQ, and T_EIDGL, represent a significant fraction of CDF in the BNL review, but not in the SNPS PRA.

(3) The T_6W sequence becomes a relatively insignificant contributor in the BNL review, although its frequency has increased slightly from that in the SNPS PRA.

(4) The ATWS sequence, $T_9C_MC_2$, one of the leading contributors in the SNPS PRA, becomes a relatively minor contributor in the BNL review.

(5) There are no counterparts to contributions from sequence types, T_RL_RQUH and $T_4C_MKUU_H$, in the SNPS PRA.

Some remarks can also be made with regard to contributions from different initiators in the BNL study[4] and the SNPS PRA[2] based on comparisons between Table A.10 and Table A.7.

(1) For Class I, the dominant initiator is LOOP, both in the BNL review and the SNPS PRA. Loss of condenser vacuum initiator becomes less important, while loss of service water initiator assumes greater prominence in the BNL review. Accident sequences initiated by dc bus failure also become relatively insignificant contributors in the BNL review.

(2) For Class II, the loss-of-condenser vacuum initiator is the leading contributor in both studies. Loss of service water emerged as a significant contributor in the BNL review to replace manual shutdown and loss of offsite power in the SNPS PRA as next in importance for Class II.

(3) For the ATWS sequence (Class IV), there is a very large increase in the BNL study of contributions from the sequences initiated by turbine trip. MSIV closure, which was the predominant contributor in the SNPS PRA, becomes less dominant in the BNL review. Also, contribution from the loss-of-feedwater ATWS becomes relatively insignificant in the BNL review.

A.1.2.3 Application of IDCOR-IPE Methodology to SNPS Core-Damage Profile

LILCO[6] has recently applied the IDCOR-IPE methodology to the SNPS resulting in a third core-damage profile. A detailed review of the relevant LILCO report is beyond the scope of this study, but the core-damage profile presented in that report is compared with that developed by the SNPS PRA and the BNL review.

The IDCOR-IPE methodology application identified a total of more than 360 core damage sequences, 61 of which have frequencies larger than 10^{-7} per reactor year. As shown in Table A.4, the total CDF was estimated to be 8.5×10^{-5} per reactor year, which lies roughly halfway between the values obtained by the SNPS PRA (5.5×10^{-5}) and the BNL review (1.4×10^{-4}).

The top 30 dominant sequences of the IDCOR-IPE methodology are listed in Table A.11 in order of their importance. These sequences have a combined total frequency of 7.2×10^{-5}/reactor year, roughly 84% of the total CDF. With the exception of the top two sequences, no single sequence contributes a large fraction of the total CDF. Five out of the top seven sequences are of the sequence type, TQUX, contributing as much as 51% of the total CDF. No such distinctive feature was found in the core-damage profile developed by either the SNPS PRA or the BNL review, although the TQUX sequence was found to be a significant contributor in both. Close examination of Tables A.5 (SNPS PRA), A.8 (BNL review), and A.11 (IDCOR-IPE) discloses that five accident sequences appear in all those of these tables. No other sequence is seen to be common to Table A.8 and Table A.11. Five additional sequences, however, are common to Table A.5 and Table A.11.

Distribution of the IPE top 30 sequences into core-damage classes and sequence-types is presented in Table A.12, which, when compared with Table A.6 (for the SNPS PRA) and Table A.9 (the BNL review), shows the following:

(1) In the IDCOR-IPE methodology application, the contribution to core damage from Class I sequences amounts to almost 90% of the combined total frequencies of the top 30 sequences, whereas Class I contributes about 62% and 64% respectively for the SNPS PRA and the BNL review.

(2) Although the most dominant sequence type in both the IDCOR-IPE and the BNL review is the T_1QUX sequence, the relative contribution is much higher in the IDCOR-IPE (66%) than in the BNL review (20%). For the SNPS PRA, the T_1QUX sequence and the ATWS sequence, $T_9C_MC_2$, each contributes about 22%, as mentioned in the previous section.

(3) In contrast with the SNPS PRA and the BNL review, the contribution from Class IV (ATWS) sequences is insignificant in the IDCOR-IPE.

(4) Unlike the SNPS PRA or the BNL review, sequences initiated by a small LOCA (S_2) appear in both Class II and Class III in the IDCOR-IPE, although their overall contribution to CDF is insignificant.

CDF contributions categorized according to accident initiators and classes are presented in Table A.13 for the IDCOR-IPE methodology application. Note that these results are based on all of the accident sequences identified in the IDCOR-IPE event tree analysis, not just on the top 30 sequences listed in Table A.11. For each of the core-damage classes, Table A.14 provides an overview of the relative importance of each contributor. The following remarks can be made by comparing Table A.14 with Table A.7 (for the SNPS PRA) and Table A.10 (for the BNL review):

(1) For Class I, there is a large increase with the IDCOR-IPE in the sequence frequency contributions associated with a turbine trip compared to the SNPS PRA or the BNL review. This makes turbine trip the top initiator of the IDCOR-IPE list, replacing loss of offsite power which was the foremost contributor in both the SNPS PRA and the BNL review. A similar comment can be made with regard to the loss of condenser vacuum initiator, which is ranked second in the IDCOR-IPE list.

(2) The accident sequences initiated by internal flooding or failures in level instrumentation and drywell cooling do not appear as a significant contributor in the IDCOR-IPE list, although they both rank near the top in the SNPS PRA and the BNL review.

(3) For Class II, accident sequences initiated by a small LOCA emerge as the top contributors in the IDCOR-IPE list instead of loss-of-condenser vacuum for SNPS PRA and BNL review. The internal flooding initiator becomes the second leading contributor, although its frequency is identical to that of the SNPS PRA. As a whole, Class II sequences make relatively insignificant contributions to the total CDF (about 6%) in Table A.14, compared to 15.5% in the SNPS PRA and 9.3% in the BNL review.

(4) The small LOCA initiator also replaces the medium LOCA initiator of SNPS PRA and BNL review to become the top contributor for Class III in the IDCOR-IPE, although its overall contribution is insignificant (about 2%).

(5) There is a significant decrease in the contribution from Class IV (ATWS) sequences in the IDCOR-IPE (6%), as compared to the SNPS PRA (24%) or the BNL review (32%). However, MSIV closure and loss-of-condenser vacuum with loss of feedwater or turbine trip initiators remain the three leading contributors for Class IV, as was the case with the SNPS PRA.

A.1.2.4 Explanations of Discrepancies in Dominant Sequence Quantifications

As noted previously, five accident sequences appeared among the top 30 dominant sequences in all of the three lists (i.e., Tables A.5, A.8, and A.11). In addition, two distinct subsets (each containing five sequences) of Table A.5 appear in Table A.8 and Table A.11 separately. In most cases, the sequence frequency estimated by the BNL reassessment or by the IDCOR-IPE methodology application differ from that obtained by the SNPS PRA. The reasons behind these quantitative differences are very briefly discussed in the following, for each of these sequences. The sequence frequencies shown without

parenthesis correspond to those evaluated by the BNL review. Those enclosed in single parenthesis are from the SNPS PRA. Those shown with double parentheses are the results of IDCOR-IPE methodology application.

(1) T_TQUX 5.5E-6 (2.4E-6) ((2.1E-5))

This sequence, one of the five sequences appearing in all three top 30 dominant sequence lists, is initiated by a turbine trip, followed by failures of feedwater and high-pressure injection (HPCI & RCIC), and the operator's failure to initiate timely reactor depressurization. The discrepancies in the sequence frequencies can be attributed to the different initiator frequencies and unavailability data used in the quantifications, as summarized in the following:

	T_T(/ry)	QU	X
BNL Review	8.0	8.2E-4	8.4E-4
SNPS PRA	4.49	6.3E-4	8.4E-4
IDCOR-IPE	6.7	1.3E-3	2.4E-3

(2) T_MQUX 2.5E-6 (7.2E-7) ((3.4E-6))

This sequence, initiated by MSIV closure, is also common to Tables A.5, A.8, and A.11. The following table shows the differences in sequence frequency quantifications.

	T_M(/ry)	QU	X
BNL Review	0.67	4.5E-3	8.4E-4
SNPS PRA	0.32	2.7E-3	8.4E-4
IDCOR-IPE	0.36	3.9E-3	2.4E-3

(3) M_SQUX 1.6E-6 (1.3E-6) ((6.2E-7))

For this sequence initiated by manual shutdown, the same initiator frequency (4.3/ry) was used in all of the three analyses. The differences in sequence frequencies stem mainly from the following unavailability data used.

	QU	X
BNL Review	4.5E-4	8.4E-4
SNPS PRA	3.5E-4	8.4E-4
IDCOR-IPE	6.0E-5	2.4E-3

(4) T_EIDUV 1.4E-6 (9.9E-7) ((6.4E-7))

The initiator frequencies and unavailability data used in quantifying this LOOP sequence are summarized as follows:

	T_E(/ry)	I	D	U	V
BNL Review	0.15	0.37	3.6E-3	1.1E-2	0.63
SNPS PRA	0.082	0.52	3.3E-3	1.1E-2	0.63
IDCOR-IPE	0.062	0.37	3.6E-3	.4E-2	0.63

For the IDCOR-IPE, the support state contribution of 6.2E-5 is added to T_EID, which is then multiplied by the complement of G (0.5), the failure probability of drywell temperature control and UV.

(5) FS_0QUX 1.0E-5 (1.7E-6) ((1.7E-6))

For this sequence which involves MSIV closure with an accumulation of greater than 3'-10" of water in the reactor building (source = other than condensate storage tank (CST)), the event tree analysis and sequence quantification performed in the IDCOR-IPE are completely identical to those in the SNPS PRA. The result of the BNL review is an estimate based on an independent study carried out at BNL.[8]

(6) T_CUX 4.2E-6 (3.1E-6)

This is one of the five accident sequence which are common only to Table A.5 (SNPS PRA) and Table A.8 (BNL review). The slightly larger sequence frequency obtained in the BNL review is essentially because of the larger initiator frequency used (0.41/ry vs 0.5/ry). The unavailability of the high pressure injection system (U) is also slightly higher (0.009 vs 0.01) in the BNL review.

(7) $T_T C_M K C_2$ 4.2E-6 (9.9E-7)

For this ATWS sequence, initiated by a turbine trip with bypass available, separate initiator frequencies are used depending on the reactor power level at the time of accident in the SNPS PRA. The high power and low power initiator frequencies are taken to be 0.85/ry and 1.3/ry, respectively. The failure probability for adequate reactivity control, C_2, is 0.11 for the high power initiator and 1.5E-2 for the low power initiator. In the BNL review, the initiator frequency is taken to be 5.3/ry, which corresponds to a high power level. Also, C_2 is estimated to be 0.15.

(8) T_DDIQ 2.2E-6 (2.2E-6)

For this sequence involving the loss of the dc bus, no change was made in the BNL reassessment.

(9) T_EIIIDV 1.7E-6 (1.5E-6)

For this sequence, initiated by loss of offsite power, there is a factor of two increase in the initiator frequency for the BNL review (8.0E-6 vs 4.3E-6). This increase, however, is offset by a smaller value (0.63 vs 0.82) assigned to III, the failure probability to recover offsite power at four hours. Also, the complement of the failure probability to maintain long term depressurization which does not appear explicitly in the accident sequence expression is lower (0.7 vs 0.95) in the BNL review.

(10) T_EIIIDUV 1.1E-6 (1.2E-6)

For this loss of offsite power sequence, the initiator frequency used in the BNL review is slightly higher. This is balanced, however, by the lower value (0.63 vs. 0.82) used for III (failure probability to recover offsite power at four hours). The complement of the failure probability to maintain

long term depressurization, which does not appear explicitly in the sequence expression, is also lower (0.8 vs 0.95). The net effect is a slight decrease in the BNL quantification.

(11) T_IQUX (6.7E-7) ((4.7E-6))

This sequence, initiated by the inadvertent opening of a relief valve, is one of the five sequences appearing only in Table A.5 (SNPS PRA) and Table A.11 (IDCOR-IPE). The sequence frequency estimated by the IDCOR-IPE methodology application is almost a factor of seven larger due mostly to the larger initiator frequency (0.15/ry vs 0.09/ry) and the larger values of X (2.4E-3 vs 8.4E-4) and QU (1.3E-2 vs 8.91E-3) used.

(12) T_EIVDUX' (2.2E-6) ((2.3E-6))

Although the sequence frequency estimated by IDCOR-IPE for this LOOP initiated sequence is very close to that obtained in the SNPS PRA, there are significant differences in the initiator frequency (2.7E-5 for the SNPS PRA vs 1.7E-5) and the failure probability of IV (0.26 for the SNPS PRA vs 0.58). The unavailability of the high-pressure injection system is also different (0.75 for the SNPS PRA vs 0.58). These differences compensate one another.

(13) T_DQUX (5.3E-7) ((6.7E-7))

The initiator of this accident sequence is the failure of the 125V dc bus (Division II), the frequency of which is taken to be 4.4E-3/ry in both the SNPS PRA and the IDCOR-IPE methodology application. The unavailability of feedwater, Q, and the probability of UX (failures of both high-pressure injection and timely reactor depressurization) are estimated to be 0.19 and 6.4E-4 respectively in the SNPS PRA. In the IDCOR-IPE methodology application, the failure probability of X is given a value of 6.6E-3, while QU is taken to be 2.3E-2.

(14) T_EIUV (7.7E-7) ((6.4E-7))

For this sequence, initiated by loss of offsite power, the slightly lower frequency obtained by the IDCOR-IPE methodology application can be explained by the following data:

	T_E(/ry)	I	U	V
SNPS-PRA	0.082	0.52	9.0E-3	2.E-3
IDCOR-IPE	0.062	0.37	1.4E-2	2.E-3

(15) T_RRQUX (1.1E-6) ((1.1E-6))

This sequence involves a reactor water level instrument line leak outside of containment. The relevant event tree analyses and sequence frequency quantifications performed in the two studies are completely identical.

A.1.2.5 Summary of SNPS Core-Damage Profiles

The following summary can be made regarding the core-damage profiles developed by the SNPS PRA,[2] the BNL review[4] and the IDCOR-IPE methodology application.[6]

(1) For the SNPS PRA and the BNL review, no single sequence was found to contribute a large fraction of the total CDF. The core-damage profile consists of a large number of accident sequences, each making a small contribution. A similar trend was found in the IDCOR-IPE methodology application, except for the top two sequences, T_TQUX and T_CQUX, which contribute about 25% and 14%, respectively.

(2) The top 30 sequences of both the SNPS PRA and the IDCOR-IPE methodology application contribute about 73% and 84% respectively to the total CDF, whereas, in the BNL review, the top 31 sequences account for about 77% of the total CDF.

(3) The following sequences are dominant both in the SNPS PRA and the BNL review: (i) loss-of-offsite power sequences, (ii) ATWS sequences because of MSIV closure, and (iii) internal flooding and level instrumentation. For the IDCOR-IPE methodology application, the dominant sequences are initiated by turbine trip, loss-of-condenser vacuum, and loss-of-offsite power.

(4) In the IDCOR-IPE methodology application, the Class I sequences contribute approximately 86% to the total CDF, and only about 58% in both the SNPS PRA and the BNL review.

(5) Contribution to the total CDF from Class IV (ATWS) sequences is significantly lower in the IDCOR-IPE methodology application (6%), than in the SNPS PRA (24%) and the BNL review (32%).

(6) The following five accident sequences were among the top 30 dominant sequences in all of the three studies: T_TQUX, T_MQUX, T_EIDUV, M_SQUX, and FS_0QUX.

(7) Contributions from loss of condenser vacuum and loss of a dc bus rank much higher in the SNPS PRA than in the BNL review, although their absolute frequency is of the same order of magnitude. For the IDCOR-IPE methodology application, the frequency for loss-of-condenser vacuum initiator is about two to four times higher than that in the SNPS PRA or the BNL review.

(8) The ATWS sequences initiated by a turbine trip rank high in the BNL review, but relatively low in the SNPS PRA and the IDCOR-IPE lists.

(9) The contribution from sequences induced by loss of service water ranks higher on the BNL review list, than in the SNPS PRA and the IDCOR-IPE lists.

(10) The sequence type, T_1QUX, is the foremost contributor to CDF in all of the three studies. T_2UX and T_8W sequences also make relatively large contributions in both the SNPS PRA and the BNL review, but not in the IDCOR-IPE methodology application.

(11) The ATWS sequence $T_9C_MC_2$ is a dominant contributor in the SNPS PRA. Its dominant position, however, is superseded by the $T_7C_MKU_H$ and T_TC_MKQ sequences in the BNL review.

A.2 Core-Meltdown Phenomena and Containment Response

In the previous section important core-meltdown accident sequences were identified in terms of the overall core-melt frequency. In this section, a review of the core-meltdown phenomena and containment response appropriate to these accident sequences is presented. In addition, accident sequences are examined which, although they do not appear to be important to the overall core-melt frequency, may pose a unique or very severe threat to containment integrity. Neither IDCOR nor SARRP has performed detailed severe-accident sequence evaluations for Mark II plants. Thus much of the core-meltdown phenomenology will be based on analyses of Mark I plants. Other studies specific to a BWR with a Mark II containment are assessed including the results of the Containment Loads Working Group (CLWG)[9] and the Containment Performance Working Group (CPWG).[10]

A typical Mark II containment building is shown in Figure A.1. The Mark II containment volumes are relatively small and rely on water to condense any steam that might be released from the primary system during an accident. Containments of this design are called pressure-suppression containments. Mark II containments are very effective at condensing steam, but their small volume makes them vulnerable to any combustible and noncondensable gases that would be generated during a severe core-meltdown accident. In order to reduce their vulnerability to combustion all Mark II containment atmospheres are continuously inerted with nitrogen during plant operation.

The aim of this section is to identify severe-accident containment loads (pressure/temperature histories) appropriate to the accident sequences identified in Section A.1. These loads are then used to determine the most probable mode of containment failure. This, in turn, identifies the potential release paths for fission products to reach the environment. This section, therefore, provides the link between the identification of core-meltdown accident sequences and the determination of fission-product release paths.

The review of the IDCOR and SARP analyses of core meltdown phenomena and containment response was greatly assisted by the IDCOR/NRC meetings that were held specifically to identify differences between the approaches adopted by the two groups and to develop a way of resolving these differences. These meetings identified 18 broad NRC/IDCOR issues that highlighted significant differences between the approaches of the two groups. These issues are listed in Table A.15, but they do not all apply to a BWR and some are not related to core-meltdown phenomena and containment response. Out of the 18 issues, 8 have been identified that are appropriate to the subject of this section. Each issue is discussed, in turn, in the following sections. Differences between IDCOR and SARP will be identified and their significance indicated.

A.2.1 In-Vessel Hydrogen Generation (NRC/IDCOR Issue 5)

There were (and still are) significant differences between the IDCOR and SARP predictions of hydrogen (H_2) generation during in-vessel core melting. During the early stages of core heatup and degradation (while the fuel rods are still in place in the core region), both IDCOR and SARP predicted similar H_2 generation. However, after the fuel rods and cladding began to melt and relocate into the bottom of the reactor vessel, the SARP analysis indicated more H2 generation than the IDCOR analysis.

Hydrogen is important to containment loading because it is a combustible and noncondensable gas. The Mark II containments are inerted with nitrogen during plant operation and consequently, H_2 combustion (Issue 17) is not a threat to the Mark II containments unless their integrity is lost and oxygen is introduced, thus deinerting the atmosphere. However, Mark II containments are small and, therefore, any significant buildup of noncondensable gases (such as H_2) could threaten containment integrity by overpressure. The larger amount of H_2 generated in-vessel in the SARP analysis resulted in a higher predicted containment pressure before vessel failure than in the IDCOR analysis. The BNL staff performed an extensive assessment[11] of in-vessel H_2 generation, particularly, with regard to accidents that resulted in core damage but which were terminated by subsequent coolant injection. The results of these calculations indicated the potential for more H_2 generation than predicted by IDCOR. However, both studies allocated a very low probability for overpressure failure of the Mark II containment from in-vessel H_2 generation. The authors concur that there is a low probability of containment failure because of H_2 accumulation before RPV failure and, therefore, the issue does not appear to significantly affect the potential for large fission-product release in Mark II containments. However, this issue is of more importance to other containment designs that are not inerted.

A.2.2 Core Slump, Core Collapse, and Reactor Vessel Failure (NRC/IDCOR Issue 6)

This is another area in which there were significant differences between the IDCOR and SARP analyses. The importance of these differences to overall risk depended on a plant specific capability to mitigate a core-melt accident. Section A.2.1 indicated that the predicted hydrogen generation during core slump was quite different in the IDCOR and SARP analyses but that the impact is not expected to be important for the Mark II containment because the atmosphere was inert.

The importance of the core slump and reactor vessel failure models is on how they influence the initial conditions for ex-vessel interactions of the core debris with water or concrete. The IDCOR core slump model assumed that after 20% of the core had melted, it relocated into the bottom of the reactor vessel, which, in turn, rapidly failed because of local penetration failure. Thus only a relatively small fraction of the core was initially released from the reactor vessel. The remainder of the core melted down over a much longer time period. A similar philosophy has been adopted in the NRC staff issue paper on direct heating.[12] This work states that the BWR core support design (which provides individual support for each group of four fuel bundles from the vessel bottom head) is judged to minimize the probability of high-pressure ejection of core debris into the containment. Slumping of relative small quantities of core debris (because of localized failure of the supports) is anticipated to result in depressurization of the vessel (because of local melt-through) before large quantities of molten core material have collected in the bottom head.

On the other hand, the Battelle Columbus Laboratories (BCL) analysis of Peach Bottom[13] with the Source Term Code Package (STCP) assumed total collapse of the core into the bottom of the reactor vessel after 75% of the core was predicted to melt. Thus, all of the core debris was available to be released when the vessel was predicted to fail in the BCL model. The much larger

quantity of core materials released from the vessel at the time of vessel failure in the BCL model has important implications for the Mark I containment. If the primary system was at high pressure during core meltdown, then the molten core materials would be ejected under pressure from the reactor vessel when it failed. In Section A.2.4, the phenomena that could occur when molten core debris is ejected from the reactor vessel under pressure are discussed. Since more core debris was predicted to be ejected (BCL model), the resulting pressure/temperature loads in containment were correspondingly higher. The Severe Accident Risk Reduction Program (SARRP) has also performed an uncertainty study, in support of NUREG-1150,[12] which examines the range of possible core slump behavior and attaches a low likelihood to the high core-melt fraction slump model.

If the primary system was depressurized during core meltdown, then the core debris would fall under gravity into the region below the reactor vessel after it failed. Some Mark II containments have downcomers in the pedestal region which would channel the molten debris into the suppression pool with a potential for severe fuel-coolant interactions. Other Mark II plants would tend to confine the core debris in the pedestal region. Thus there would be a potential for vigorous core-concrete interaction and significant fission product vaporization.

From the above discussion, it is clear that differences between the IDCOR and SARP models for core slump and vessel failure are significant and do have an important influence on the potential for early containment failure.

This is an area of significant uncertainty with very little experimental evidence to support either the IDCOR or SARP models. Both models are credible and seem to indicate the range of possible outcomes of a core-meltdown accident.

A.2.3 Containment Failure Because of In-Vessel Steam Explosions (Issue 7)

The potential for an in-vessel steam explosion to occur and generate a missile capable of failing containment was investigated by a group of experts and the results were published in NUREG-1116.[14] The conclusion of this expert group was that the occurrence of such an event has a relatively low probability. The allocation of a very low conditional probability (10^{-4}) of occurrence to this event appears to be well supported by the existing data base.

A.2.4 Direct Heating of Containment (Issue 8)

This was identified in the SARP analysis as an area of significant phenomenological uncertainty related specifically to core meltdown with the primary system at high pressure. If molten core materials were to be ejected from the reactor vessel under pressure, experiments[15] at Sandia National Laboratories (SNL) have indicated that they could form fine aerosols, which might be dispersed into the containment atmosphere and directly heat it. An additional concern was the oxidation of the metallic content of the core debris. These reactions are very exothermic and would add an additional heat load to the containment. Exothermic chemical reactions were less of a concern in Mark II containments because they are inerted (low oxygen content). However, the zirconium-steam reaction could still contribute to containment loading.

The pressure rise in containment because of direct heating is directly proportional to the quantity of core debris dispersed from the reactor vessel. Section A.2.2 noted that the BCL analysis predicted significantly more debris release at vessel failure than the IDCOR analyses predicted. Thus, the potential for early containment failure because of direct heating was considered in the SARP analysis, but it was not considered credible in the IDCOR analysis. The assumption that all the core debris is released at vessel failure (SARP analysis) is clearly conservative (SARP assumption) with respect to containment loading. However, the IDCOR results may be too optimistic considering the lack of supporting large-scale experiments.

Therefore, given the present state of phenomenological uncertainty associated with this issue and the existence of relatively simple mitigative solutions, the following points are offered. There are two ways of potentially mitigating the effects of a high-pressure meltdown. The first is to convert a high-pressure sequence into a low-pressure sequence by ensuring that the ADS is activated. Note that venting may also be required for some sequences (e.g., station blackout) to ensure that the containment pressure does not increase beyond the relief valve capability. The second way to mitigate a high-pressure meltdown is by drywell sprays. The drywell spray, flooding the drywell floor, may cool the core debris and reduce the potential for the core to attack the diaphragm floor and downcomers. In addition to cooling the core debris, the drywell sprays will aid in decontaminating the drywell atmosphere and may substantially reduce the released fission products even for cases with drywell failure.

A.2.5 Ex-Vessel Heat Transfer Model from Molten Core to Concrete (Issue 10)

This issue is of concern to Mark II containments because heat transfer from the top of molten core materials (on the drywell floor) directly heats the drywell atmosphere. Differences in heat transfer from the top of the core debris resulted in significant differences in the predicted drywell atmospheric pressures and temperatures for Peach Bottom and these differences are expected to persist for Mark II plants. The IDCOR model transferred more heat from the top of the core debris than the SARP model. Thus, IDCOR predicted much higher drywell temperatures than the SARP analyses.

Differences in the predicted drywell pressure/temperature histories influenced the potential for suppression pool bypass (Issue 13A) and containment performance (Issue 15).

A.2.6 Suppression Pool Bypass (Issue 13A)

If the fission products pass through the suppression pool, both IDCOR and SARP predicted significant retention of fission product aerosols in the water. The amount of retention depended on several factors such as submergence, water temperature, aerosol particle size, carrier gas composition, and others. The ability of the Mark II suppression pool to trap aerosol fission products was found to be an important mitigative feature. Thus, any pathways that might open, which would allow the fission products to bypass the pool are very undesirable. The following are possible ways in which the suppression pool may be bypassed:

- loss of drywell isolation
- failure of vacuum breakers between the drywell and wetwell
- failure of drywell penetrations because of high temperature
- structural failure of the drywell because of high pressure
- failure of the drywell floor or downcomers as a result of contact with molten core materials

Because of the importance of the suppression pool as a mitigative feature, the vulnerability of a Mark II containment to any of the above bypass pathways was carefully assessed. The authors support the low probabilities given in the SARP analysis for loss of drywell isolation and failure of the vacuum breakers. The degradation of the drywell penetrations because of high temperatures in the SARP analysis was based on the work of the CPWG, which had significant BNL input.

A.2.7 Containment Performance (Issue 15)

The response of a Mark II containment to severe accident loads is uncertain. It was noted in Section A.2.5 that IDCOR predicted relatively high drywell temperatures. Seal degradation was assumed to result in a gradual leakage from the drywell in the SARP analysis. This was again based on the work by the CPWG.[10]

Differences in containment performance can influence the timing and quantities of fission products released to the reactor building (refer to Section A.3). However, for the Mark II containment both models tend to predict late failure and gradual leakage.

A.2.8 Secondary Containment Performance (Issue 16)

The secondary containment (reactor building) surrounds the primary containment and has the potential to trap fission products released during a severe accident. For the Mark I containment, there were differences between the IDCOR and SARP analyses with regard to hydrogen combustion in the secondary containment. These differences are expected to result in significant differences in the amount of fission products predicted to be retained in the secondary containment for the Mark II containment as well.

A.3 Fission-Product Release

Section A.2 identified potential containment failure modes or fission-product release paths appropriate to the important core-meltdown accident sequences identified in Section A.1. The aim of this section is to determine the timing and amount of fission products released from the damaged fuel and predict the subsequent reduction of these fission products along the release paths identified in Section A.2. Since current calculations for a Mark II plant were not available, the IDCOR and SARP analyses for the Peach Bottom plant were used as the basis for these assessments.

In order to review differences in approach the IDCOR and NRC contractor analyses (performed for SARP) are compared in Tables A.16 and A.17 for ATWS and station blackout sequences in a Mark I containment. The IDCOR methods predicted slightly higher releases of the more volatile fission-product groups (iodine and cesium) whereas the SARP methods predicted much higher releases of

the less-volatile fission-product groups (strontium, lanthanum, etc.). The reasons for the different predictions in Tables A.16 and A.17 are complex but were identified during the numerous IDCOR/NRC meetings and they are included in the list of 18 NRC/IDCOR issues in Table A.15. Out of the 18 issues, 6 are pertinent to fission-product release and transport. However, not all of the 6 are major contributors to the differences in Tables A.16 and A.17. Each of the 6 issues are discussed in the following subsections.

A.3.1 Fission-Product Release Before Vessel Failure (Issue 1)

This is one issue that does not contribute significantly to the differences between the IDCOR and SARP analyses in Tables A.16 and A.17. Both studies predicted similar releases of the more volatile fission products during in-vessel core degradation with the exception of tellurium (Te). However, a recent report by IDCOR assessed the impact of Te treatment and modeled similar in-vessel Te releases to the SARP analyses. Differences in the predicted environmental releases of Te in Tables A.16 and A.17 are, therefore, not because of differences in the in-vessel Te release and retention models, but are due to differences in the amount of retention predicted to occur in the secondary containment (refer to Section A.3.6).

A.3.2 Fission-Product and Aerosol Retention in the Primary System (Issue 4)

Differences in the initial primary system retention predicted by IDCOR and SARP were again not too significant and differ by less than a factor of two. The important difference between the IDCOR and SARP models was that in the SARP analysis, fission products retained in the primary system at the point of vessel failure were permanently retained, whereas in the IDCOR analysis revaporization of these fission products after vessel failure was modeled. This is discussed in more detail in Section A.3.4.

A.3.3 Ex-Vessel Fission-Product Release (Issue 9)

There were significant differences between the IDCOR and SARP analyses for fission-product release as a result of core-concrete interactions. The higher releases of the strontium, lanthanum and cesium groups in Tables A.16 and A.17 in the SARP analyses were because of the modeling of ex-vessel fission-product release. The potential for fission-product release and inert aerosol generation during core-concrete interactions was not modeled in the IDCOR analysis of Peach Bottom.[16] IDCOR argued that by modeling the aerosol generation during core-concrete interactions, the increased aerosol density in containment would increase aerosol agglomeration and settling, thus reducing the predicted environmental release fractions relative to those predicted without this additional aerosol source. This IDCOR position is clearly indefensible.

In addition, the IDCOR predicted core debris temperatures during core-concrete interactions were high and, based on experimental evidence, one would expect the release of some of the refractory fission-product groups at these temperatures. Therefore, the authors believe that IDCOR should calculate the release of the refractory fission products and the associated inert aerosols.

A.3.4 Revaporization of Fission Products from the Primary System (Issue 11)

Section A.3.2 indicated that revaporization was an area of major difference between the IDCOR and SARP analyses. Recall that SARP does not model revaporization of fission products from the primary system after reactor vessel failure whereas IDCOR does model this effect. To observe the influence of the different approaches the distribution of cesium-iodide (CsI) is tracked at various stages of a station blackout sequence in Peach Bottom (Table A.18).

At the point of vessel failure, both IDCOR and the NRC contractors' analyses performed for SARP indicated significant primary system retention of CsI (100% retention for IDCOR and 85% retention for SARP). Immediately before containment failure, the NRC contractor analysis has 85% of the CsI permanently retained in the primary system, whereas the IDCOR model has revaporized 24% of the CsI from the primary system. At the end of the calculation, the IDCOR model has revaporized 91% of the CsI.

The IDCOR revaporization model means that significantly more of the volatile fission products were predicted to be released to the reactor building than in the NRC contractor approach in which revaporization was not modeled. For the sequence under consideration in Table A.18, the environmental releases were similar despite the major differences discussed above. The reason for the similar environmental releases was because of the much greater retention of fission products in the reactor building predicted in the IDCOR model vs. the SARP model. This is discussed in more detail in Section A.3.6.

A.3.5 Fission-Product Deposition Model in Containment (Issue 12)

This was another issue that does not contribute significantly to the differences between the IDCOR and SARP analyses in Tables A.16 and A.17. Issues 9, 11, and 16 appear to drive the differences in these two analyses. However, this issue may be of more importance to other containment designs.

A.3.6 Secondary Containment Performance (Issue 16)

Section A.2.8 indicated that secondary containment (reactor building) performance was an area of major difference between the IDCOR and SARP analyses. The extent of the difference can be seen from Table A.18, which shows that IDCOR predicts that only 6% of the CsI entering the reactor building would be released to the environment compared with 40% released in the NRC contractor analysis.

The above differences were significant and were found to be because of several factors. For example, the gas flowrate through the secondary containment was higher in the SARP analysis, which allows less time for aerosol settling. In addition, IDCOR models natural circulation paths in the secondary containment, which further increased the residence times. Such paths were not modeled in the SARP analysis. Finally, SARP calculated several hydrogen burns to occur in the secondary containment, which rapidly blew out the atmosphere of the building. IDCOR calculated more gradual combustion phenomena, which did not result in rapid blowout of the secondary containment atmosphere.

There was uncertainty with regard to how much retention of aerosol fission products would occur in the secondary containment. Even in the SARP

analysis, there was the potential for significant retention. However, given the uncertainty, the secondary containment should not be relied on as the only means of mitigating fission-product releases.

A.4 Offsite Consequences

In this section, the potential offsite consequences of the severe accidents described in the previous sections are examined. There is one NRC/IDCOR issue related to offsite consequences, which concerned differences in the assumed evacuation models. Differences in the evacuation model influence the predicted early health effects. The issue was largely resolved and was related to the fraction of the population assumed not to participate in the evacuation. Neither IDCOR nor SARP have performed consequence analyses for the Mark II containment. So the Peach Bottom results are used to obtain "insights" as to the effects of the new source term methods on suppression pool containments. The containment loads[9] and performance[10] analysis and the BNL review of Limerick[3] are used to obtain containment performance and release path insights specific to the Mark II containment.

Table A.19 gives the person-rem calculations from IDCOR[17] and NUREG-1150 for several accident sequences and failure modes in Peach Bottom. This table indicates that if the containment is predicted to fail (either early or late) and the suppression pool is bypassed, then the offsite person-rem predictions are similar (within the range 0.1 to $3x10^7$) for the accidents considered. The only time that a significant reduction (to $4x10^5$) in person-rem was calculated, was with successful wetwell venting and no pool bypass. These results clearly show that wetwell venting and the prevention of pool bypass are the keys to mitigating the fission product releases for a Mark I containment.

The estimated releases for the various accident classes are given in Table A.20 for the Limerick plant. Note that the release calculations are based on RSS methodology and are inconsistent with the present IDCOR and SARP methods. The main point of the comparison is that the most severe releases are expected to occur for early containment failure (Class IV) with pool bypass. The γ'' failure mode (wetwell failure below the pool level) is assumed to drain the pool with no fission-product scrubbing. The Class IV γ failure also results in substantial releases because of the fission product release during core-concrete interaction (since this release is unscrubbed). Even the very late containment failure events (e.g., Class II γ) result in substantial release fractions for the volatile fission products. The net result is shown in Figures A.3 and A.4. BNL predicted substantially higher risk for Limerick (on a point estimate basis) than the RSS predicted for Peach Bottom. It is concluded that even though containment failure is generally delayed (see CLWG report[9]) for the Mark II containment because of its larger volume compared to the Mark I containment, the delay does not have a major effect on the releases. Thus, as with the Mark I containment, it remains true for the Mark II containment, that pool bypass conditions should be avoided.

A.5 Summary and Risk Insights

A.5.1 Core-Damage Profile

As has been observed by others for BWRs, transients rather than LOCAs dominated the core-damage risk profile for the studies examined in Section

A.1. Otherwise, there was no consistent pattern of relative ranking of transient sequences among the studies. For the two Mark II plants examined in Section A.1, there appeared to be many more sequences which contributed a significant fraction to core melt. However, the accident classes still exhibit the same general trends found for the Mark I plant. Specifically transients and LOCAs with make-up failure (including blackout related) appear to dominate the CDF but loss of CHR heat removal and ATWS sequences also contribute.

For the Limerick and Shoreham review, TQUV/TQUX sequences were important contributors to core melt. Most of this contribution was because of a high failure rate for primary system depressurization. Improved ADS reliability is addressed in Section 4.3.3.

It is important to recognize that the qualitative accident sequence descriptors are rather general and that different hardware and/or operational failures in the various Mark II plants could lead to the same general accident sequence. In order to identify the plant-specific (and often unique) vulnerabilities that contribute to a given general sequence descriptor (e.g., station blackout) in a given plant, a plant-specific examination (such as a failure mode and effects analysis coupled with a fault tree/event tree analysis or an equivalent method) would be needed.

A.5.2 Consequence Analysis

The assessment of core meltdown phenomena and containment response indicated that the Mark II containment is somewhat less vulnerable to severe-accident containment loads than the Mark I plant. However, unless mitigative actions are taken, a Mark II containment has the potential to fail a few hours after the reactor vessel fails. If containment failure occurs in the drywell, any fission products in the drywell atmosphere could pass to the reactor building without the benefit of suppression pool scrubbing. Because of this vulnerability, the predicted offsite consequences appeared to be relatively insensitive to the accident sequence definition. For Class I and II accidents the containment failure is expected in the drywell and a large release of non-volatile fission products is expected. For Class IV accidents there is a possibility of failing below the pool and draining it with essentially no scrubbing.

A.6 References

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16. "Peach Bottom Atomic Power Station-Integrated Containment Analyses," IDCOR Technical Report T23.1PB, March 1985.

17. "Technical Support for Issue Resolution," IDCOR Technical Report T85.2, July 1986.

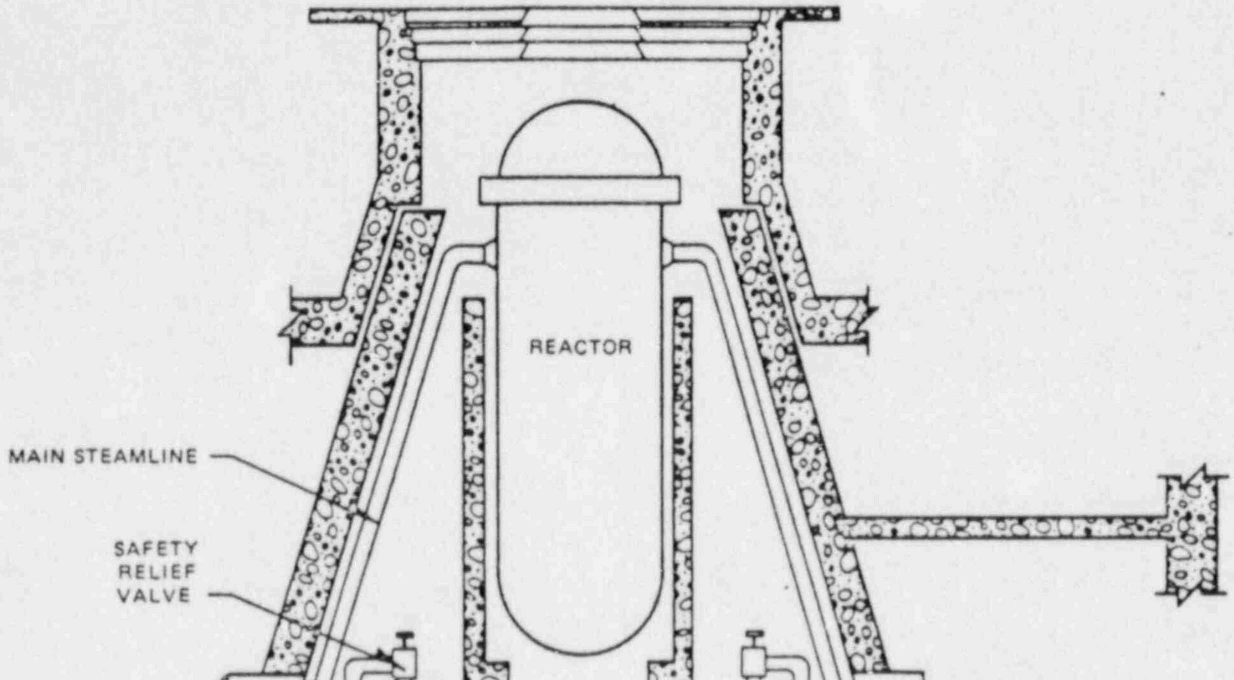

Figure A.1 Typical Mark II pressure suppression containment.

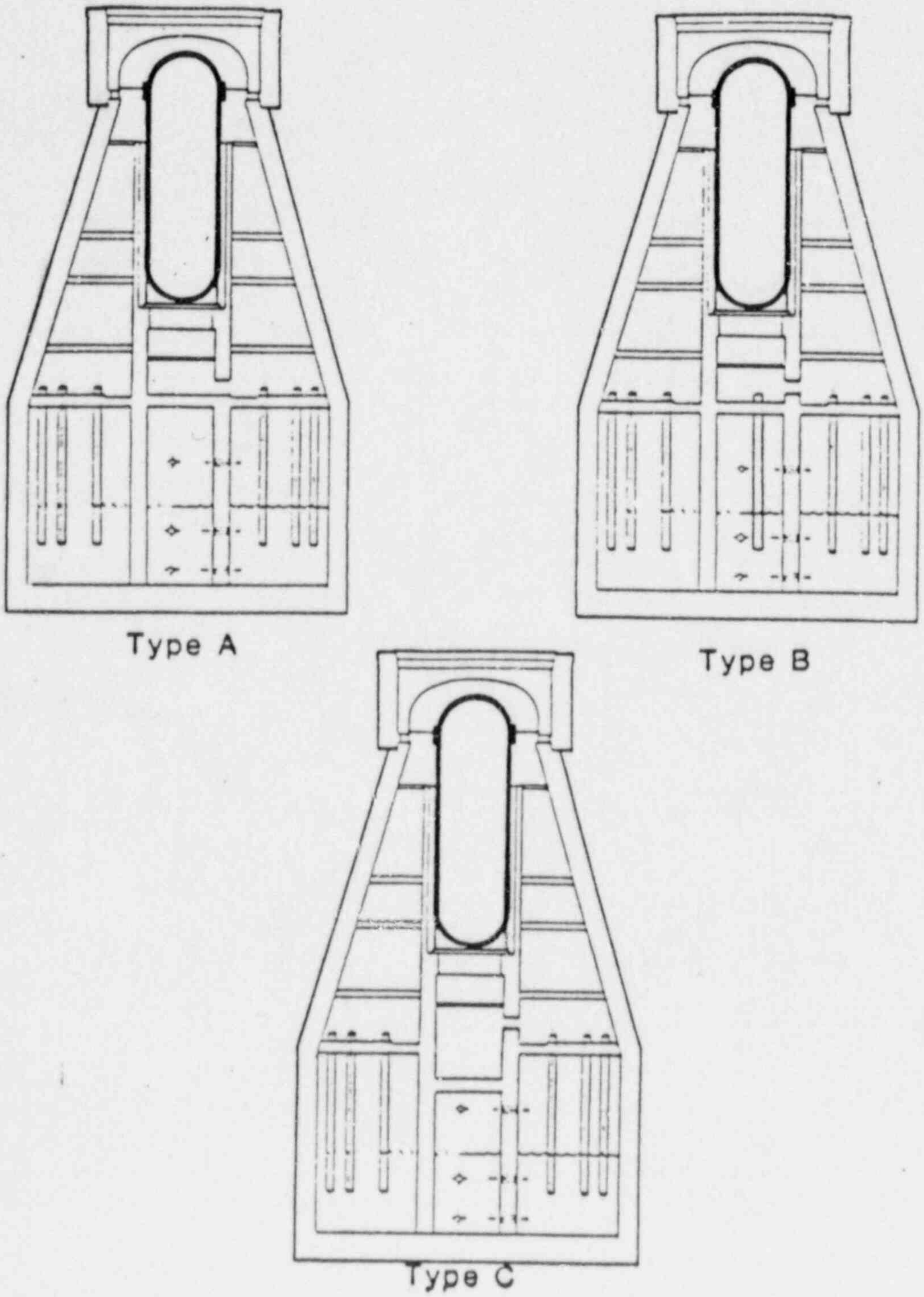

Figure A.2 Typical Mark II containment designs.

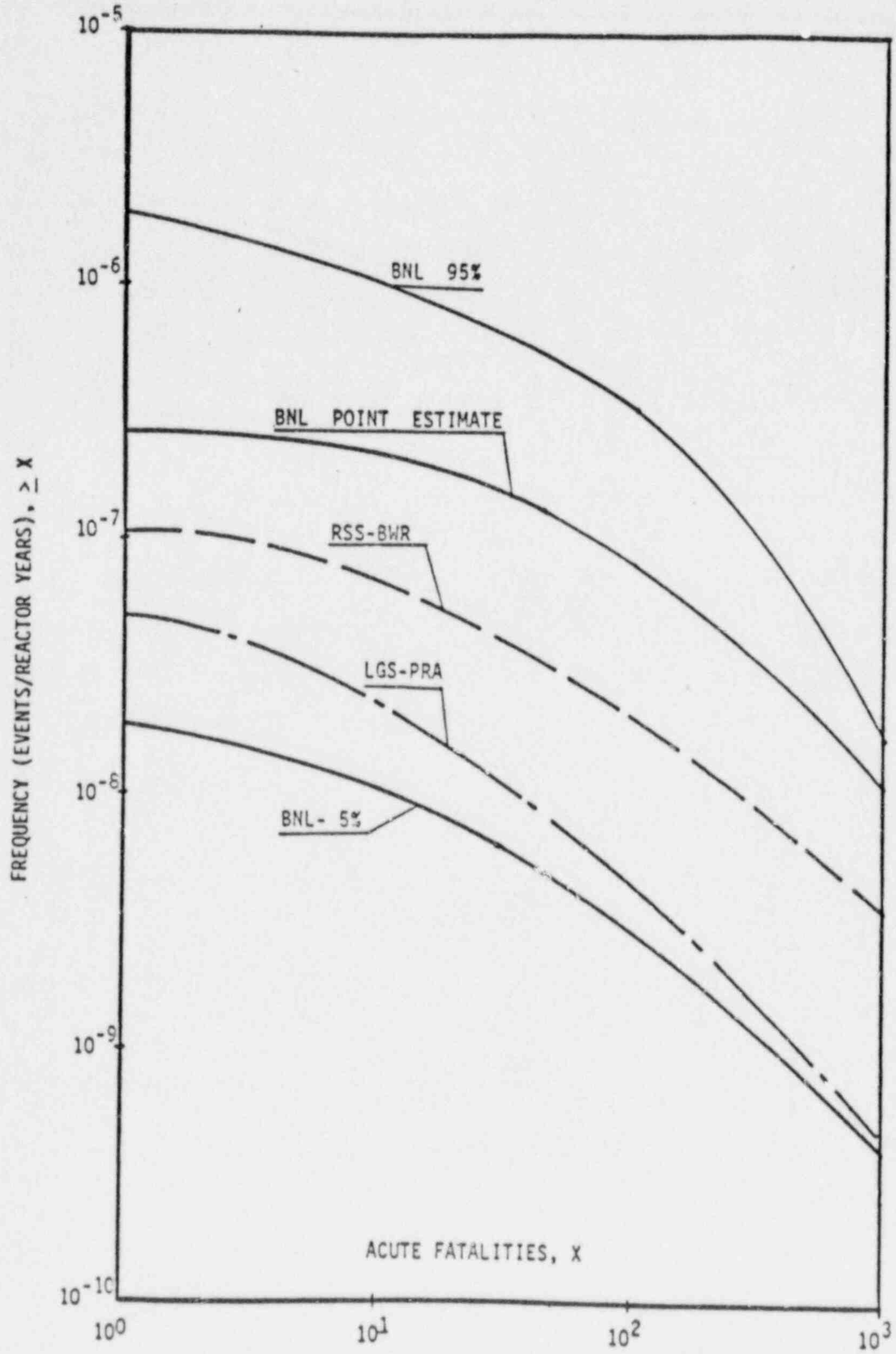

Figure A.3 Estimated complementary cumulative distribution functions for acute fatalities for Limerick.

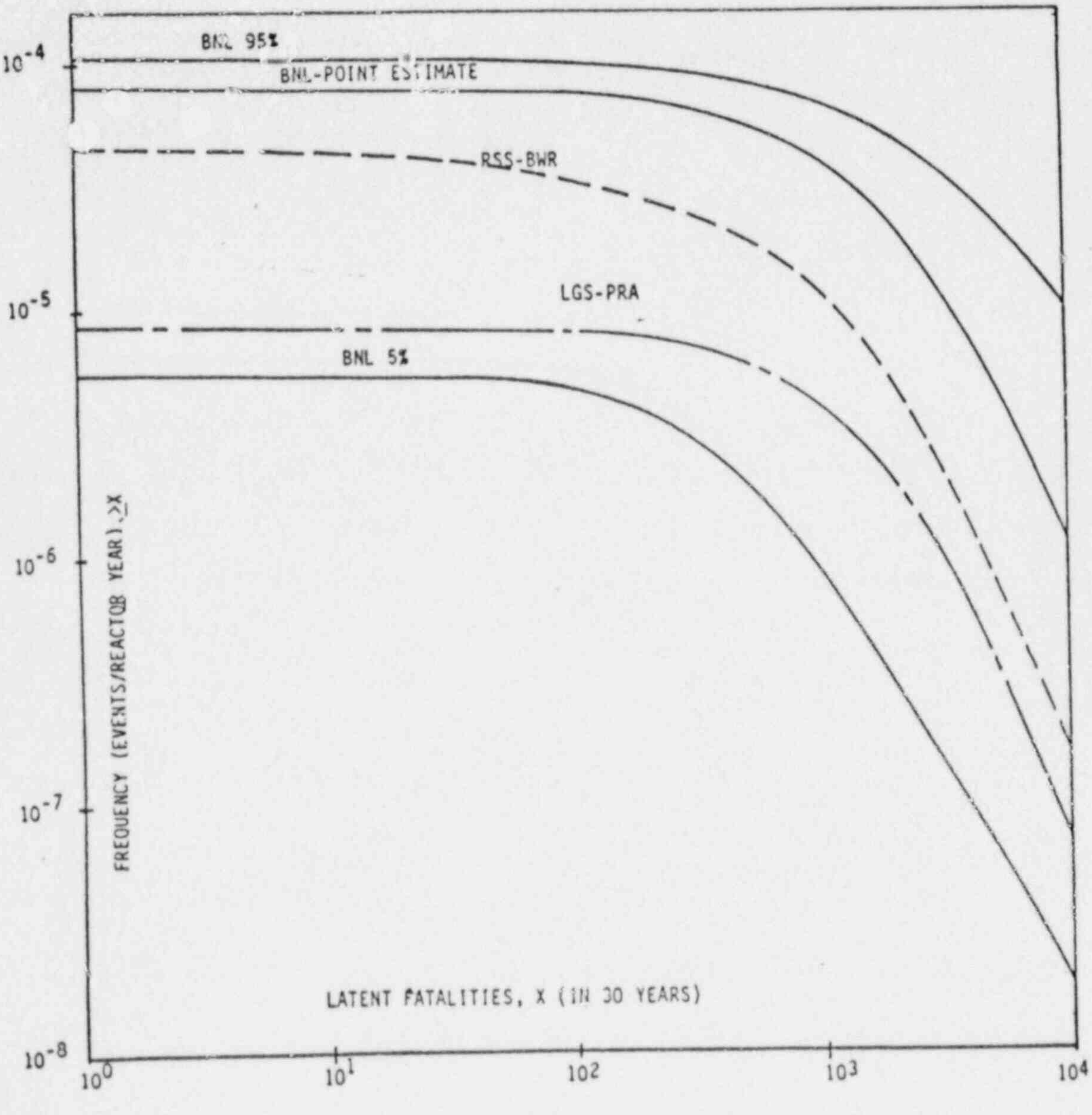

Figure A.4 Estimated complementary cumulative distribution functions for latent fatalities for Limerick.

Glossary for Table A.1, A.2, A.5, A.8 and A.11

A	Large LOCA
AC	Loss of ac power
C'	Timely scram initiation
C_M	Mechanical failure to scram
C_2	Adequate reactivity control through poison injection
D	Recovery of Division I or II diesel generator (at different times), or ADS not inadvertently actuated for ATWS sequences (Table A.5)
DC	Loss of dc power
DI	125 V dc bus failure, Division I
FS_0	MSIV closure with accumulation of water in reactor building
G	Drywell heat removal
H	Operator recognizes need for injection
I	Recovery of offsite power at 30 minutes (time phase I)
II	Recovery of offsite power at two hours (time phase II)
III or III_A	Recovery of offsite power at four hours (time phase III A)
III_B	Recovery of offsite power at four hours (time phase III B)
III_C	Recovery of offsite power at four hours (time phase III C)
IV	Recovery of offsite power at ten hours (time phase IV)
J	Containment heat removal failure
K	ARI (Alternate rod insertion)
L	Level control and stable cooling established
L_{R1}	Opposite division low level trip electronic
L_{R2}	Opposite division low level trip electronic
M_S	Manual shutdown
O	Flashing detected
O_R	Operator error caused leak in alternate reference leg
P	Safety relief valve recloses
Q	Feedwater system
R	Continued power operation
S	MSIVs open, PCS available
S_0	MSIV closure and flood from source other than CST
S_2	Small LOCA
S_C	MSIV closure and flood from CST via HPCI or RCIC
T	Successful crosstie of turbine building service water for reactor building
T_C	Loss of condenser vacuum
T_D	Loss of 125V dc bus (Division II)
T_{D2}	Loss of multiple dc buses
T_E, or T_{E3}	Loss of offsite power
T_F	Loss of feedwater
T_I	Inadvertent opening of relief valve transient
T_M	MSIV closure transient
T_{MC}	MSIV closure/loss of condenser vacuum
T_{M2}	MSIV closure transient (ATWS)
T_R	Reactor water level instrument line leak outside containment
T_{SW}	Loss of reactor building service water
T_T	Turbine trip
T_{T1}	Turbine trip with bypass and with feedwater
U	High pressure injection function
U'	RCIC (reactor core isolation cooling) system

Glossary for Table A.1, A.2, A.5, A.8 and A.11 (Cont'd)

U"	HPCI (nigh pressure core injection) system
U_H	Operator failure to control level 1
U_R	Failure of RCIC
V	Low pressure injection function
W or W_d	Containment heat removal function (includes residual heat removal system and power conversion system)
W'	RHR or RCIC in steam condensing mode
W"	Power conversion system
W_{12}	Failure of one RHR
WSW	Loss of service water
X	Depressurization (via automatic depressurization system or manual)
X'	Maintain depressurization (long term)

Table A.1 Ranking of Limerick Dominant Core-Damage Sequences by LGS PRA (Total Core-Damage Frequency = 1.5×10^{-5}/reactor year)

Rank	Core-Damage Sequence	Frequency (per reactor year)	Class	% of Total Core-Damage Frequency**
1	$T_E UV$	5.97E-6†	I	39.8
2	$T_F QUX$	3.63E-6	I	24.2
3	$T_T QUX$	7.73E-7	I	5.2
4	$T_E UX$	6.9E-7	I	4.6
5	$T_I UX$	6.8E-7	I	4.5
6	$T_T PW$	3.87E-7	II	2.6
7	$T_{T1} C_M PU$	2.70E-7	III	1.8
8	$T_F C_M UU_R$	2.36E-7	III	1.6
9	$T_M QUX$	2.20E-7	I	1.5
10	$T_F C_M W_{12}$	1.85E-7	IV	1.2
11	$T_F C_M PU$	1.64E-7	III	1.1
12	$T_F QW$	1.55E-7	II	1.0
13	$T_E C_M W_{12}$	1.43E-7	III/IV*	0.95
				------- 90%
14	$T_I CUX$	1.40E-7	I	0.93
15	$T_F QUV$	1.40E-7	I	0.93
16	$T_F PW$	1.40E-7	II	0.93
17	$T_T C_M C_2$	1.01E-7	III/IV*	0.67
18	$T_I W$	6.83E-8	II	0.46
19	AJ	6.40E-8	II	0.43
20	$T_I C_M U$	6.37E-8	III	0.42

*80% Class III and 20% Class IV.
**Class I 81.66%, Class II 5.42%, Class III 5.68%, Class IV 2.06%.
†5.97E-6 = 5.97×10^{-6}.

Table A.2 Ranking of Limerick Dominant Core-Damage Sequences by BNL Review (Total Core-Damage Frequency = 1.0×10^{-4}/reactor year)

Rank		Core-Damage Frequency	Frequency (per reactor year)		Class	% of Total Core-Damage Frequency
BNL Review	LGS PRA		BNL Review	LGS PRA		
1	2	$T_F QUX$	3.7E-5	3.63E-6	I	37.
2	1	$T_E UV$	3.2E-5	5.97E-6	I	32.
3	4	$T_E UX$	8.6E-6	6.9E-7	I	8.6
4	3	$T_T QUX$	8.0E-6	7.73E-7	I	8.0
5	5	$T_I UX$	4.0E-6	6.8E-7	I	4.0
6	-	$T_T(DC)$	2.0E-6	---	I	2.0
						-------90%
7	12	$T_F QW$	1.3E-6	1.55E-7	II	1.3
8	-	$T_T(WSW)$	1.2E-6	---	I	1.2
9	15	$T_F QUV$	1.1E-6	1.4E-7	I	1.1
						-------95%
10	7	$T_{T1} C_M PU$	8.7E-7	2.7E-7	III	0.87
11	6	$T_T PW$	7.7E-7	3.87E-7	II	0.77
12	25	$T_E W$	6.4E-7	3.51E-8	II	0.64
13	-	$T_T(AC)$	6.1E-7	---	I	0.61
14	8	$T_F C_M UU_R$	5.3E-7	2.36E-7	III	0.53
15	14	$T_I C'UX$	5.0E-7	1.4E-7	I	0.50
16	18	$T_I W$	4.3E-7	6.83E-8	II	0.43
17	13	$T_{E3} C_M W_{12}$	4.3E-7	1.43E-7	III	0.43
18	23	$T_I UV$	3.6E-7	2.6E-8	I	0.36
19	9	$T_M QUX$	3.6E-7	2.2E-7	I	0.36
20	-	$T_F(DC)$	3.1E-7	---	I	0.31

*Class I 96%, Class II 3.1%, Class III 1.8%, Class IV 0%.

Table A.3 Definition of Accident Sequence Subclasses

Core-Damage Class	Subclass	Definition
Class I	A	Accident sequences involving makeup in which the reactor pressure remains high
	B	Accident sequences involving a loss of offsite power and loss of coolant inventory makeup
	C	Accident sequences involving a loss of coolant inventory induced by an ATWS sequence
	D	Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced to 200 psi.; accident sequences initiated by common mode failures disabling multiple systems (ECCS) leading to loss of coolant inventory makeup
	E	Accident sequences caused by common mode failures which result in multiple frontline system failures with the reactor at high pressure
Class III	A	Accident sequences leading to core vulnerable conditions initiated by vessel rupture where the containment integrity is not breached in the initial time phase of the accident
	B	Accident sequences initiated or resulting in small or medium LOCAs for which the reactor cannot be depressurized
	C	Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is at low pressure
	D	Accident sequences which are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate, challenging the containment integrity

Table A.4 Comparison of Core-Damage Frequencies Obtained by SNPS PRA, BNL Review and IDCOR-IPE Methodology Application

Accident Sequence Initiator		Core-Damage (CD) Class					Core-Damage Frequency (per yr)	% of Total CDF
		I	II	III	IV	V		
Transients (turbine trip, MSIV closure, manual shut-down & other)	SNPS	8.7E-6	4.8E-6				1.3E-5	23.64
	BNL	1.5E-5	6.4E-6				2.2E-5	15.7
	IDCOR	4.3E-5	1.9E-6	1.1E-7			4.5E-5	52.9
Loss of off-site ac power (LOOP)	SNPS	9.9E-6	1.1E-6				1.1E-5	20.0
	BNL	2.9E-5	1.4E-6				3.0E-5	21.4
	IDCOR	1.1E-5	9.6E-8				1.1E-5	12.9
ATWS (Anticipated Transient Without Scram)	SNPS	4.0E-6		2.1E-9	1.4E-5		1.8E-5	32.7
	BNL	*		2.8E-8	4.5E-5		4.5E-5	32.1
	IDCOR	9.6E-6		5.6E-9	4.9E-6		1.5E-5	17.6
LOCA (Loss of Coolant Accidents)	SNPS		1.0E-6	1.0E-6			2.0E-6	3.6
	BNL		5.3E-7	1.3E-6			1.8E-6	1.3
	IDCOR	6.9E-7	1.5E-6	1.5E-6	7.0E-9		3.7E-6	4.4
Level Instrumentation (reference leg & drywell cooling)	SNPS	3.8E-6	1.2E-7	5.2E-9			3.9E-6	7.1
	BNL	1.2E-5	2.5E-8	1.5E-7			1.2E-5	8.6
	IDCOR	2.5E-6	1.2E-7		1.9E-7		2.8E-6	3.3
Flooding at Elevation 8 of Reactor Bldg.	SNPS	3.1E-6	7.8E-7				3.9E-6	7.1
	BNL	1.8E-5	2.0E-6				2.0E-5	14.3
	IDCOR	3.1E-6	7.8E-7		1.7E-10		3.9E-6	4.6
LOCA Outside Drywell	SNPS					3.7E-8	3.7E-8	0.67
	BNL					2.0E-7	2.0E-7	0.14
	IDCOR		4.5E-9			9.0E-8	9.5E-8	0.11
Loss of Service Water or DC Bus	SNPS	3.0E-6	7.7E-7				3.8E-6	6.9
	BNL	7.6E-6	2.4E-6				1.0E-5	7.1
	IDCOR	3.0E-6	5.5E-7		6.9E-8		3.6E-6	4.2
Total	SNPS	3.2E-5	8.5E-6	1.0E-6	1.4E-5	3.7E-8	5.5E-5	
	BNL	8.2E-5	1.3E-5	1.5E-6	4.5E-5	4.2E-7	1.4E-4	
	IDCOR	7.3E-5	5.0E-6	1.6E-6	5.2E-6	9.0E-8	8.5E-5	

*In the BNL review, all ATWS sequences are assumed to lead to core damage Class IV, based partly on the judgement that the operator will not be able to inhibit ADS.

Table A.5 Dominant Accident Sequence Frequencies by SNPS PRA

Rank	Sequence	Frequency* (per reactor year)	Class	% of Total CDF
1	$T_{M2}C_MC_2$	6.4(-6)	IV	11.64
2	T_CUX	3.1(-6)	IA	5.64
3	T_TQUX	2.4(-6)	IA	4.36
4	T_DDIQ	2.2(-6)	IA	4.0
5	T_EIVDUX'	2.2(-6)	IB	4.0
6	FS_0QUX	1.7(-6)	ID	3.09
7	T_EIII_CDV	1.5(-6)	IB	2.73
8	T_FC_MU	1.5(-6)	IC	2.73
9	T_FC_MUD	1.5(-6)	IV	2.73
10	$T_CW'W''$	1.5(-6)	II	2.73
11	M_SQUX	1.3(-6)	IA	2.36
12	T_EIII_ADUV	1.2(-6)	IB	2.18
13	T_EW_d	1.1(-6)	II	2.0
14	T_RRQUX	1.1(-6)	IA	2.0
15	$T_FC_MC_2$	1.0(-6)	IV	1.82
16	T_EIDUV	9.9(-7)	ID	1.8
17	$T_{T1}C_MC_2$	9.9(-7)	IV	1.8
18	$T_TW'W''$	8.9(-7)	II	1.62
19	T_EGOL	8.4(-7)	ID	1.53
20	$M_SW'W''$	8.2(-7)	II	1.49
21	T_EIUV	7.7(-7)	ID	1.4
22	T_MQUX	7.2(-7)	IA	1.31
23	T_IQUX	6.7(-7)	IA	1.22
24	$T_EIIDU''V$	6.5(-7)	IB	1.18
25	$T_{M2}C_MC_2U$	6.2(-7)	IC	1.13
26	$T_{M2}C_MC_2UD$	6.2(-7)	IV	1.13
27	$T_EC_MC_2$	6.0(-7)	IV	1.09
28	FS_CQUX	5.5(-7)	ID	1.0
29	T_DQUX	5.3(-7)	IA	0.96
30	$T_{T1}C_MU$	5.3(-7)	IC	0.96
		Total = 4.05×10^{-5}		Sum = 73.63%

*a(-b) means $a\times10^{-6}$.

Table A.6 Distribution of Core-Damage Frequency Contributions According to Sequence Types (SNPS PRA)

Core-Damage Class	Subclass	Sequence Type or Sequence*	Frequency (per reactor year)	Percentage
Class I	A	T_1QUX	9.0E-6	22.15
		T_CUX	3.1E-6	7.66
		T_DDIQ	2.2E-6	5.43
	B	T_2DUX'	2.2E-6	5.43
		T_3DUV	1.85E-6	4.57
		T_4DV	1.5E-6	3.71
	C	T_5C_MU	2.0E-6	5.01
		$T_6C_MC_2U$	6.2E-7	1.53
	D	T_7UV	1.76E-6	4.35
		T_EGOL	8.4E-7	2.08
			2.51E-5	61.92
Class II		T_8W	4.3E-6	10.65
Class IV		$T_9C_MC_2$	9.0E-6	22.2
		$T_{10}UD$	2.1E-6	5.24
			1.11E-5	27.44
			4.05E-5	100.

*$T_1 = T_T$, M_S, T_RR, FS_0 etc.
$T_2 = T_EIV$
$T_3 = T_EII$, T_EIII_A
$T_4 = T_EIII_C$
$T_5 = T_F$, T_{T1}
$T_6 = T_{M2}$
$T_7 = T_EI$, T_EID
$T_8 = T_C$, T_T, M_S, T_E
$T_9 = T_F$, T_{T1}, T_{M2}, T_E
$T_{10} = T_FC_M$, $T_{M2}C_MC_2$

Table A.7 Distribution of Core-Damage Frequency (CDF) Contributions According to Initiators (SNPS PRA)

Class	Accident Initiator	Frequency (per reactor year)	Percentage
Class I (58.2% of total CDF)	Loss of offsite power	9.9E-6	30.9
	Level instrumentation and drywell cooling failure	3.8E-6	11.9
	Internal flooding	3.1E-6	9.7
	Loss of condenser vacuum	3.0E-6	9.4
	DC bus failure	2.7E-6	8.4
	Turbine trip	2.5E-6	7.8
	Loss of feedwater ATWS	1.8E-6	5.6
	Manual shutdown	1.4E-6	4.4
	Turbine trip ATWS	1.2E-6	3.8
	Others	2.6E-6	8.1
		3.2E-5	100.0
Class II (15.45% of total CDF)	Loss of condenser vacuum	2.1E-6	24.7
	Manual shutdown	1.2E-6	14.1
	Loss of offsite power	1.1E-6	12.9
	Turbine trip	1.0E-6	11.8
	Internal flooding	7.8E-7	9.2
	Others	2.32E-6	27.3
		8.5E-6	100.0
Class III (1.8% of total CDF)	Medium LOCA	4.9E-7	49.0
	RPV LOCA	3.1E-7	31.0
	Large LOCA	1.8E-7	18.0
	Small LOCA	1.6E-8	1.6
	Others	4.0E-9	0.4
		1.0E-6	100.0
Class IV (all ATWS, 24.4% of total CDF)	MSIV closure/loss of condenser vacuum	7.4E-6	55.2
	Loss of feedwater	2.8E-6	20.9
	Turbine trip	2.3E-6	17.2
	Loss of offsite power	6.9E-7	5.2
	Others	2.0E-7	1.5
		1.34E-5	100.0
Class V (0.067% of total CDF)	LOCA outside drywell	3.7E-8	100.0

Total CDF = 5.5E-5

Table A.8 Dominant Accident Sequence Frequencies for SNPS by BNL Review

Rank	Sequence	Frequency*	Class	% of Total CDF
1	$T_T C_M KQ$	1.0(-5)	IV	7.07
2	$T_E IDGL$	1.0(-5)	IB	7.07
3	$FS_0 QUX$	1.0(-5)**	IA	7.07
4	$T_M C_M KU_H$	8.3(-6)	IV	5.87
5	$T_T C_M KU_H$	6.7(-6)	IV	4.73
6	$T_E IVD$	6.7(-6)	IB	4.73
7	$T_T QUX$	5.5(-6)	IA	3.89
8	$T_T C_M KC_2$	4.2(-6)	IV	2.97
9	$T_C UX$	4.2(-6)	IA	2.97
10	$T_T C_M KUU_H$	3.9(-6)	IV	2.76
11	$T_E IIIDUX'$	3.3(-6)	IB	2.33
12	$T_{SW} TSUV$	2.6(-6)	ID	1.84
13	$T_{SW} TSUX$	2.6(-6)	IA	1.84
14	$T_M QUX$	2.5(-6)	IA	1.77
15	$T_C W$	2.5(-6)	II	1.77
16	$T_T C_M KW$	2.4(-6)	IV	1.70
17	$T_R RL_{R1} QUH$	2.4(-6)	IA	1.70
18	$T_R RL_{R2} QUH$	2.2(-6)	IA	1.55
19	$T_D D_I Q$	2.2(-6)	IA	1.55
20	$T_R RO_R QUX$	2.0(-6)	IA	1.41
21	$T_E IGL$	1.9(-6)	IA	1.34
22	$T_E IIIDV$	1.7(-6)	IB	1.2
23	$M_S QUX$	1.6(-6)	IA	1.13
24	$T_{SW} TSW$	1.4(-6)	II	0.99
25	$T_E IDUV$	1.4(-6)	ID	0.99
26	$T_E IIVW$	1.4(-6)	II	0.99
27	$T_E IIIDX$	1.2(-6)	IB	0.85
28	$T_T C_M KPQ$	1.1(-6)	IV	0.78
29	$T_E C_M KU_H$	1.1(-6)	IV	0.78
30	$T_E IIIDUV$	1.1(-6)	IB	0.78
31	$T_M C_M KUU_H$	1.0(-6)	IV	0.71

*a(-5) means $a \times 10^{-5}$.
**Estimated by BNL review.

Total = 1.09×10^{-4} Sum = 77.13%

Table A.9 Distribution of Core-Damage Frequency Contributions According to Sequence Types (BNL Review of SNPS PRA)

Core-Damage Class	Subclass	Sequence Type or Sequence*	Frequency (per reactor year)	Percentage
Class I	A	T_1QUX	2.16E-5	19.8
		T_2UX	6.8E-6	6.23
		T_RL_RQUH	4.6E-6	4.22
		T_DDIQ	2.2E-6	2.02
		T_EIGL	1.9E-6	1.74
	B	T_3DUX'	3.3E-6	3.03
		T_3DV	1.7E-6	1.56
		T_3DUV	1.1E-6	1.01
	Others (T_3DX, T_EIVD, T_EIDGL)		1.79E-5	16.41
	C	$T_4C_MKUU_H$	4.9E-6	4.49
	D	T_5UV	4.0E-6	3.67
			7.0E-5	64.16
Class II		T_6W	5.3E-6	4.86
Class IV		$T_7C_MKU_H$	1.61E-5	14.76
		T_TC_MKQ, T_TC_MKPQ	1.11E-5	10.17
		$T_TC_MKC_2$	4.2E-6	3.85
		T_TC_MKW	2.4E-6	2.2
			3.38E-5	30.98
			1.09E-4	100.0

*T_1 = T_T, T_M, M_S, FS_0 etc.
T_2 = T_C, $T_{SW}TS$
T_3 = T_EIII
T_4 = T_T, T_M
T_5 = T_EID, $T_{SW}TS$
T_6 = T_C, $T_{SW}TS$, T_EIIV
T_7 = T_M, T_T, T_E

Table A.10 Distribution of Core-Damage Frequency Contributions According to Initiators (BNL Review of SNPS PRA)

Class	Accident Initiator	Frequency (per yr)	Percentage
Class I (58.3% of total CDF)	Loss of offsite power	2.9E-5	35.5
	Internal flooding	1.8E-5	22.1
	Level instrumentation and drywell cooling failure	1.2E-5	14.7
	Turbine trip	5.2E-6	6.4
	Loss of service water	5.2E-6	6.4
	Loss of condenser vacuum	4.8E-6	5.9
	MSIV closure	2.7E-6	3.3
	DC bus failure	2.4E-6	2.9
	Manual shutdown	1.8E-6	2.2
	Others	5.0E-7	0.6
		8.16E-5	100.0
Class II (9.3% of total CDF)	Loss of condenser vacuum	3.4E-6	26.2
	Loss of service water	2.4E-6	18.5
	Internal flooding	2.0E-6	15.4
	Turbine trip	1.5E-6	11.5
	Loss of offsite power	1.4E-6	10.8
	Others	2.3E-6	17.7
		1.3E-5	100.0
Class III (1.07% of total CDF)	Medium LOCA	6.1E-7	40.7
	Large LOCA	3.7E-7	24.7
	RPV LOCA	3.1E-7	20.7
	Others	2.1E-7	14.0
		1.5E-6	100.0
Class IV (all ATWS, 31.9% of total CDF)	Turbine trip	2.9E-5	64.9
	MSIV closure/loss of condenser vacuum	1.1E-5	24.6
	Loss of feedwater	2.6E-6	5.8
	Loss of offsite power	1.4E-6	3.1
	Others	7.0E-7	1.6
		4.47E-5	100.0
Class V (0.14% of total C.D.F.)	LOCA outside drywell	2.0E-7	100.0

Total CDF = 1.4E-4

Table A.11 Dominant Accident Sequence Frequencies for SNPS by IDCOR-IPE Methodology Application

Rank	Sequence	Frequency* (per reactor year)	Class	% of Total CDF
1	T_TQUX	2.1(-5)	IA	24.71
2	T_CQUX	1.2(-5)	IA	14.12
3	T_IQUX	4.7(-6)	IA	5.53
4	T_FC_MUV	3.6(-6)	IC	4.24
5	T_MQUX	3.4(-6)	IA	4.0
6	T_EIVDUX'	2.3(-6)	IB	2.71
7	$T_{D2}QUX$	2.2(-6)	IA	2.59
8	T_EIII_BDGUX	1.9(-6)	IB	2.24
9	FS_0QUX	1.7(-6)	ID	2.0
10	T_TC_MUV	1.6(-6)	IC	1.88
11	$T_{MC}C_MC_2WUV$	1.6(-6)	IV	1.88
12	$T_FC_MC_2WUV$	1.2(-6)	IV	1.41
13	T_RRQUX	1.1(-6)	IA	1.29
14	T_EIVDX'	1.1(-6)	IB	1.29
15	T_EIDGUX	1.0(-6)	IB	1.18
16	$T_TC_MC_2WUV$	1.0(-6)	IV	1.18
17	$T_{MC}C_MUV$	1.0(-6)	IC	1.18
18	T_IC_MUV	9.4(-7)	IC	1.11
19	T_I[illegible]	9.0(-7)	ID	1.06
20	T_EIUX	7.7(-7)	IA	0.91
21	S_2GQUV	7.2(-7)	II	0.85
22	S_2WQUV	7.2(-7)	II	0.85
23	S_2QUX	7.0(-7)	IIIB	0.82
24	T_CWQUV	6.8(-7)	II	0.8
25	T_DQUX	6.7(-7)	IA	0.79
26	T_EIUV	6.4(-7)	IB	0.75
27	T_EIDUV	6.4(-7)	IB	0.75
28	M_SQUX	6.2(-7)	IA	0.73
29	M_SWQUV	6.0(-7)	II	0.71
30	$T_{MC}C_MC_2UV$	6.0(-7)	IC	0.71

*a(-5) means a x 10^{-5} Total = 7.16×10^{-5} Sum = 84.27%

Table A.12 Distribution of Core-Damage Frequency Contributions According to Sequence Types (SNPS IDCOR-IPE Methodology Application)

Core-Damage Class	Subclass	Sequence Type or Sequence*	Frequency (per reactor year)	Percentage
Class I	A	T_1QUX	4.74E-5	66.2
		T_2UX	7.7E-7	1.08
	B	T_3DGUX	2.9E-6	4.05
		T_4DUX'	2.3E-6	3.21
		T_4DX'	1.1E-6	1.54
		T_2DUV	6.4E-7	0.89
		T_2UV	6.4E-7	0.89
	C	$T_5C_MC_2UV$	7.14E-6	9.97
		$T_{MC}C_MC_2UV$	6.0E-7	0.84
	D	T_IV	9.0E-7	1.26
			6.44E-5	89.93
Class II		T_6WQUV	2.0E-6	2.79
		S_2GQUV	7.2E-7	1.0
			2.72E-6	3.79
Class III	B	S_2QUX	7.0E-7	0.98
Class IV		$T_7C_MC_2WUV$	3.8E-6	5.31
			7.16E-5	100.0

*$T_1 = T_T$, T_C, FS_0 etc.
$T_2 = T_EI$
$T_3 = T_EI$, T_EIII_B
$T_4 = T_EIV$
$T_5 = T_F$, T_T etc.
$T_6 = S_2$, T_C, M_S
$T_7 = T_{MC}$, T_F, T_T

Table A.13 Summary of Core-Damage Frequency Contributions by Initiators and Classes (SNPS IDCOR-IPE Methodology Application)

Accident Sequence Initiator	Core-Damage (CD) Class I	II	III	IV	V	Sequence Totals
Transients:						
Turbine trip	2.1E-5	1.5E-7	-----	-----	-----	2.1E-5
Manual shutdown	6.2E-7	6.1E-7	-----	-----	-----	1.2E-6
MSIV closure	3.4E-6	1.8E-7	-----	-----	-----	3.6E-6
Loss of feedwater	4.3E-7	1.5E-8	-----	-----	-----	4.5E-7
Loss of condenser vacuum	1.2E-5	6.9E-7	-----	-----	-----	1.3E-5
IORV and SCRV	5.7E-6	2.4E-7	1.1E-7	-----	-----	6.1E-6
	4.3E-5	1.9E-6	1.1E-7			4.5E-5
Loss of Offsite Power	1.1E-5	9.6E-8	-----	-----	-----	1.1E-5
ATWS:						
Turbine trip	2.0E-6	-----	1.3E-9	1.1E-6	-----	3.1E-6
MSIV closure/loss of condenser vacuum	1.6E-6	-----	1.0E-9	1.8E-6	-----	3.4E-6
Loss of offsite power	1.1E-7	-----	-----	8.6E-8	-----	2.0E-7
IORV	1.4E-6	-----	-----	4.3E-7	-----	1.8E-6
Loss of FW	4.5E-6	-----	3.3E-9	1.5E-6	-----	6.0E-6
	9.6E-6		5.6E-9	4.9E-6		1.5E-5
LOCA:						
Large LOCA	-----	9.2E-8	1.3E-7	7.0E-9	-----	2.3E-7
Medium LOCA	-----	3.6E-8	3.9E-7	-----	-----	4.3E-7
Small LOCA	6.9E-7	1.4E-6	7.0E-7	-----	-----	2.8E-6
Reactor pressure vessel LOCA	-----	-----	3.0E-7	-----	-----	3.0E-7
	6.9E-7	1.5E-6	1.5E-6	7.0E-9		3.8E-6
Other Transients:						
Level instrumentation	2.4E-6	1.2E-7	-----	1.9E-7	-----	2.7E-6
Drywell cooling failure	1.4E-7	-----	-----	-----	-----	1.4E-7
Internal flooding	3.1E-6	7.8E-7	-----	1.7E-10	-----	3.9E-6
LOCA outside drywell	-----	4.5E-9	-----	-----	9.0E-8	9.5E-8
Loss of service water	2.1E-8	5.3E-7	-----	-----	-----	5.5E-7
Loss of dc bus	3.0E-6	1.8E-8	-----	6.9E-8	-----	3.1E-6
Total	7.3E-5	5.0E-6	1.6E-6	5.2E-6	9.0E-8	8.5E-5

Table A.14 Distribution of Core-Damage Frequency Contributions According to Initiators (SNPS IDCOR-IPE Methodology Application)

Class	Accident Initiator	Frequency (per reactor year)	Percentage
Class I (85.9% of total CDF)	Turbine trip	2.1E-5	28.8
	Loss of condenser vacuum	1.2E-5	16.4
	Loss of offsite power	1.1E-5	15.1
	IORV and SORV	5.7E-6	7.8
	Loss of feedwater ATWS	4.5E-6	6.2
	MSIV closure	3.4E-6	4.7
	Internal flooding	3.1E-6	4.2
	DC bus failure	3.0E-6	4.1
	Others	9.3E-6	12.8
		7.3E-5	100.0
Class II (5.9% of total CDF)	Small LOCA	1.4E-6	28.
	Internal flooding	7.8E-7	15.6
	Loss of condenser vacuum	6.9E-7	13.8
	Manual shutdown	6.1E-7	12.2
	IORV and SORV	2.4E-7	4.8
	Others	1.28E-6	25.6
		5.0E-6	100.0
Class III (1.9% of total CDF)	Small LOCA	7.0E-7	43.8
	Medium LOCA	3.9E-7	24.4
	RPV LOCA	3.0E-7	18.8
	Large LOCA	1.[illegible]-7	8.1
	Others	8.0E-8	5.0
		1.6E-6	100.0
Class IV (all ATWS, 6.1% of total CDF)	MSIV closure/loss of condenser vacuum	1.8E-6	34.6
	Loss of feedwater	1.5E-6	28.8
	Turbine trip	1.1E-6	21.2
	IORV and SORV	4.3E-7	8.3
	Others	3.7E-7	7.1
		5.2E-6	100.0
Class V (0.106% of total CDF)	LOCA outside containment	9.0E-8	100.0

Total CDF = 8.5E-5

Table A.15 NRC/IDCOR Issues

Issue	Subject
1	Fission-Product Release Before Vessel Failure
2	Recirculation of Coolant in Reactor Vessel
3	Release Model of Control Rod Materials
4	Fission Product and Aerosol Retention in the Primary System
5	In-Vessel H_2 Generation
6	Core Slump, Core Collapse, and Reactor Vessel Failure
7	Containment Failure Because of In-Vessel Steam Explosions
8	Direct Heating of Containment
9	Ex-Vessel Fission Product Release
10	Ex-Vessel Heat Transfer Model from Molten Core to Concrete
11	Revaporization of Fission Products from the Primary System
12	Fission Product Deposition Model in Containment
13A	Suppression Pool Bypass (Pool Scrubbing)
13B	Retention of Fission Products in Ice Beds
14	Modeling of Emergency Response
15	Containment Performance
16	Secondary Containment Performance
17	Hydrogen Ignition and Burning
18	Essential Equipment Performance

Table A.16 Comparison of IDCOR and SARP Predictions of Fission-Product Release for an ATWS Sequence With No Operator Actions Taken

Event	IDCOR*	NRC Contractors**
Containment Failure (hr)	1.4	1.4
Start of Core Melt (hr)	3.0	2.2
Vessel Failure (hr)	3.9	3.8
Fission-Product Release Fractions***:		
Xe-Kr	1.0	1.0
I-Br	0.1	0.03
Cs-Rb	0.1	0.03
Te-Sb	0.1	0.30
Sr-Ba	0.0004	0.43
Ru-Mo	0.001	Neg.
La	--	0.01
Ce	--	0.02

*IDCOR Technical Report 23.1PB, March 1985.
**Informal BCL Report Dated November 18, 1985.
***Fraction of Initial Core Inventory.

Table A.17 Comparison of IDCOR and SARP Predictions of Fission-Product Release for an SBO Sequence

Event	IDCOR*	NRC Contractors**
Loss of Injection (hr)	6.0	6.0
Start of Core Melt (hr)	11.4	10.7
Vessel Failure (hr)	12.0	12.2
Containment Failure (hr)	18.0	15.2
Fission-Product Release Fractions***:		
Xe-Kr	1.0	1.0
I-Br	0.05	0.012
Cs-Rb	0.05	0.014
Te-Sb	0.06	0.22
Sr	Neg.	0.37
Ru-Mo	0.0001	Neg.
La	--	0.03
Ce	--	0.05
Ba	--	0.28

*IDCOR Technical Report 23.1PB, March 1985.
**NUREG/CR-4624, Vol. 1.
***Fraction of Initial Core Inventory.

Table A.18 Comparison of IDCOR and SARP Predictions of CsI Distribution for an SBO Sequence (Fraction of Initial Core Inventory)

Location	Events					
	Vessel Failure		Containment Failure		End of Calculation	
	IDCOR* (12 hr)	NRC** (12 hr)	IDCOR (18 hr)	NRC (15 hr)	IDCOR (60 hr)	NRC (22 hr)
Reactor Pressure Vessel	1.0	0.85	0.76	0.85	0.09	0.85
Drywell	--	0.12 Melt	0.20	***	--	0.09
Suppression Pool	--	***	0.04	***	0.04	0.03
Reactor Building	--	--	--	--	0.82	0.009
Environment	--	--	--	--	0.05	0.014

*IDCOR Technical Report 23.1PB, March 1985.
**NUREG/CR-4624, Vol. 1.
***The location of the remaining fraction of CsI was not reported in the Battelle Columbus Laboratories (BCL) report.

Table A.19 Comparison of IDCOR and SARP Consequence Results (Person-Rem)

Accident Sequence	Containment Failure Mode	IDCOR	NUREG-1150
ATWS	Wetwell venting no pool bypass	3.5×10^5	--
ATWS	Wetwell venting with late pool bypass	1.0×10^7	--
Station Blackout	Containment failure at RPV failure	--	0.2 to 3.3×10^7
Station Blackout	Containment failure after a few hours	1.3×10^7	0.1 to 1.2×10^7

Table A.20 Comparison of Release Parameters from BNL Review and the Limerick PRA
(Fraction of Initial Core Inventory)

	Class I - γ			Class II - γ		Class III - γ		Class IV - γ		Class IV - γ'		Class IV - γ''	
	BNL*	BNL**	PRA	BNL	PRA	BNL	PRA	BNL	PRA	BNL	PRA	BNL	PRA
Xe-Kr	.939	.939	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
OI	0.0	0.0	0.0	0.0	.007	.007	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I	0.049	.0093	.11	.156	.06	.122	.04	.154	.261	.098	.07	.708	.73
Cs	0.055	.020	.09	.258	.023	.0542	.024	.749	.202	.749	.09	.749	.70
Te	0.058	.046	.016	.421	.4	.185	.073	.747	.434	.757	.20	.757	.55
Ba	0.006	.0017	.01	.027	.0063	.00361	.0027	.0859	.029	.0859	.016	.0859	.09
Ru	0.004	.0030	.003	.070	.069	.0169	.0086	.110	.095	.11	.088	.11	.12
La	0.00074	.00061	.0003	.0054	.0047	.00238	.00091	.0103	.00523	.0103	.006	.0103	.007

*With suppression pool flashing at containment failure.
**Without suppression pool flashing at containment failure.

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VOLUME 3

ASSESSMENT OF SEVERE ACCIDENT PREVENTION AND MITIGATION FEATURES:

BWR, MARK III CONTAINMENT DESIGN

Prepared by
R. G. Fitzpatrick, K. R. Perkins, W. J. Luckas, J. R. Lehner and W. T. Pratt

Date Published - November 1987

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Upton, New York 11973

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ABSTRACT

Guidelines and criteria have been developed for preventing and mitigating severe accidents in a BWR which has a Mark III containment (BWR Mark III). The guidelines were developed from insights derived from reviews of in-depth risk assessments performed specifically for the Grand Gulf plant and from assessments of other relevant studies. Accident sequences that dominate the core-damage frequency and those accident sequences that are of potentially high consequence were identified. Vulnerabilities of the BWR Mark III to severe accident containment loads were also identified. In addition, those features of a BWR Mark III, which are important for preventing core damage and are available for mitigating fission-product release to the environment were also identified. These guidelines and criteria are issued to provide direction to an analyst examining an individual plant. This direction calls attention to plant features and operator actions and provides the standards for assessing those features and actions found to be helpful in reducing the overall risk for Grand Gulf and other Mark III plants. Thus, the guidance is offered as a resource in examining the subject plant to determine if the same, or similar, guidelines will be of value in reducing overall plant risk. These guidelines and criteria are intended to serve solely as guidance.

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ACKNOWLEDGMENTS

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NOMENCLATURE

A	large loss of coolant accident (LOCA)
ac	alternating current
ADS	automatic depressurization system
ARI	alternate rod insertion
ASEP	Accident Sequence Evaluation Program
ATWS	anticipated transients without scram
BCL	Battelle Columbus Laboratories
BNL	Brookhaven National Laboratory
BWR	boiling water reactor
C	failure of reactor protection system (RPS)
CDF	core-damage frequency
CHR	containment heat removal
CLWG	Containment Loads Working Group
CPWG	Containment Performance Working Group
CRD	control rod drive system
dc	direct current
DG	diesel generators
E	failure of coolant injection
ECC	emergency core cooling
ECCS	emergency core cooling systems
EPG	Emergency Procedures Guidelines
GESSAR	General Electric Standard Safety Analysis Report
GI	generic issue
HEP	human error probability
HPCI	high-pressure coolant injection system
HPCS	high pressure core spray
I	failure of containment heat removal
IDCOR	Industry Degraded Core Rulemaking Program
IPE	individual plant examamination
ISL	interfacing system LOCA
LLRT	local leak rate testing
LOCA	loss-of-coolant accident
LOOP	loss of offsite power (sometimes denoted by LOSP)
LPCI	low-pressure coolant injection system
LWR	light water reactor
MCC	motor control center
MSIV	main steam isolation valve
NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
NRC/RES	U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research
P	one or more stuck open relief valves
PRA	probabilistic risk assessment
PWR	pressurized water reactor
Q	failure of feedwater system
RB	reactor building
RCIC	reactor core isolation cooling system
RPV	reactor pressure vessel
RSSMAP	Reactor Safety Study Methodology Application Program
SARP	Severe Accident Research Program
SARRP	Severe Accident Risk Reduction Program
SBO	station blackout

NOMENCLATURE (Cont'd)

SI	safety injection
SLCS	standby liquid control system
SNL	Sandia National Laboratories
SNPS	Shoreham Nuclear Power Station
STCP	Source Term Code Package
T	transient
TB	station blackout sequence (sometimes referred to as SBO)
TC	anticipated transients without scram
T_1	loss of offsite power (LOOP) initiator
T_{23}	all other transient initiators except LOOP
TPQI	transient sequence with stuck open safety relief valve failure, feedwater failure and loss of CHR
TQU	transient sequence in which the feedwater and high pressure injection systems fail
TQUX	transient sequence in which the feedwater and high pressure injection systems fail and depressurization does not occur
U	failure of high pressure injection function
USI	unresolved safety issue
V	failure of low pressure injection function
W	failure of containment heat removal (CHR)
X	failure of reactor pressure vessel (RPV) depressurization

1. EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) has formulated an approach for a systematic safety examination of existing plants to determine whether particular severe accident vulnerabilities are present and what changes are desirable to ensure that there is no undue risk to public health and safety.

The Industry Degraded Core Rulemaking Program (IDCOR) selected four reference plants for detailed analysis: Peach Bottom, Grand Gulf, Sequoyah, and Zion. The IDCOR analyses performed for the reference plants have been documented together with the methodology used for the analyses and the technical basis supporting the methodology.

Parallel with the IDCOR work, the NRC under the Severe Accident Research Program (SARP), performed risk assessments, audit calculations, sensitivity studies, and uncertainty analyses for five plants. The five plants considered by SARP were Peach Bottom, Grand Gulf, Sequoyah, Zion and Surry.

The purpose of this effort is to review all of the IDCOR and SARP analyses performed for the reference plants, understand the reasons for the differences, and then use the experienc gained from these reviews for developing guidelines and criteria that identify plant features and operator actions that were found to be important for either preventing or mitigating severe accidents in each plant type. In turn, these guidelines should prove helpful in the systematic safety examination of individual plants.

Three basic objectives or goals for this severe accident program, apply equally to all plant types:

- Goal 1: Mitigate fission-product releases.
- Goal 2: Control the frequency of high-consequence sequences.
- Goal 3: Reduce high core-damage frequency.

The aim was, therefore, to develop detailed guidelines and criteria that could be used to achieve these goals during the examination of individual plants.

"Guidelines," as used in this report, identify plant features and operator actions that were found to be important to either preventing or mitigating severe accidents in the reference plant studies. These guidelines provide a list of plant features and operator actions that the utilities can use as part of the individual plant examinations (IPEs). It is not the intent of this report to specify a set of improvements for either the reference plant or for any other plant which would be sufficient to achieve a certain level of safety. Instead, the guidelines indicate potential improvements in various areas of plant design and operation of which utilities should be aware when conducting the IPEs and making decisions on plant improvements. The intent is to provide guidance to the analyst performing an IPE, as to the plant features and operator actions which were found to reduce overall risk. It is prudent to check whether potential improvements identified in studies of other similar plants can be of help in improving overall plant performance. The guidelines contained in this report, therefore, complement the IPEs.

"Criteria," as used in this report, are the attributes which have been identified as important to assess the performance of plant features and operator actions identified in the guidelines. The criteria provide deterministic (as opposed to probabilistic) performance measures which are judged to be helpful to implement the concepts contained in the guidelines. When a decision is made to provide an item or an action specified in a guideline, the utility should address a set of questions relating to the design, operation and availability of the needed equipment and the training of operators. For the example of containment-venting guidance, the capacity of the venting system, the selection of setpoints to initiate venting, the availability of applicable procedures and the accessibility of certain valves by operators should be addressed. The criteria on containment venting provide helpful information in assessing venting capability in each individual plant.

Based on an extensive review of prior severe accident investigations, the authors have provided a set of guidelines and associated criteria which can be used to assess the capability of individual boiling water reactor (BWR), Mark III plants to cope with severe accidents. Although much of the work is based on probabilistic risk assessments (PRAs), the guidelines and criteria are deterministic in nature. That is the criteria describe specific features of key systems and operational procedures which have been found helpful in reducing the likelihood of severe accidents. The guidelines and criteria take into account detailed severe accident experiments and analyses performed by the NRC/RES, the nuclear power industry and foreign governments.

The following sections present the insights gained from reviewing the PRAs. Specifically, the IDCOR Grand Gulf Integrated Containment Analysis[1] and the SARP Grand Gulf reports[2-4] were reviewed in detail. In addition, the results of other relevant BWR PRAs were also taken into account, namely, the Grand Gulf risk assessment[5] performed as part of the Reactor Safety Study Methodology Applications Program (RSSMAP) and the PRA[6] for the General Electric Standard Safety Analysis Report (GESSAR).

1.1 Core-Damage Profile

Transients rather than loss-of-coolant accidents (LOCAs) dominated the core-damage risk profile for the PRAs reviewed. In addition, there appeared to be no pattern of relative ranking of transient sequences among the PRAs reviewed. However, in the later studies the same few functional accident sequences figured prominently in the core-damage frequency (CDF) profiles. It is also important to observe that for a given accident sequence, the major contributor to differences in quantitative results between the studies was because of subjective modeling assumptions rather than because of plant differences or data differences.

For the Grand Gulf RSSMAP study, loss of containment heat removal sequences appeared as important contributors to the CDF (about 50%). Although the more recent Accident Sequence Evaluation Program (ASEP) studies have reduced the CDF, attributable to these sequences, based on operating procedures for alternate injection, criteria have been developed to ensure that these sequences are not dominant for other BWR Mark III plants.

Both SARP and IDCOR indicated that station blackout (SBO) and anticipated transients without scram (ATWS) are the dominant core-damage sequences for the

Grand Gulf plant. Both studies calculate a total CDF only slightly higher than 10^{-6} per reactor year. Thus, the guidelines for other BWR Mark III plants will attempt to ensure that the likelihood of these "dominant" sequences are also kept at a correspondingly low level.

1.2 Consequence Analysis

The assessment of core-meltdown phenomena and containment response in the PRAs indicated that the Mark III containment is vulnerable to severe accident containment loads. Unless mitigative actions are taken, a Mark III containment has the potential to fail a short time (in a few hours or less) after the core debris melts through the reactor vessel. Therefore, fission products released into the drywell will still pass through the suppression pool. Thus, even with a containment failure, containment function (reducing the source term) is preserved for almost all cases. Only direct bypass sequences (interfacing systems LOCA) or loss of drywell integrity result in severe releases of fission products.

1.3 Guidelines and Criteria

Each guideline is provided with a detailed test of criteria which provide helpful information to assess the performance of plant features and operator actions identified in the guidelines.

1.3.1 Mitigate Fission-Product Releases

The assessment of core-meltdown phenomena and containment response indicates that the Mark III containment provides a vigorous defense against fission product release even under severe accident loads. The ability of the Mark III suppression pool to trap aerosol fission products is an important mitigative feature since it leads to a direct reduction in offsite consequences by a factor of 10 or more. Therefore, no guidelines were believed to be necessary for the Mark III design with respect to this goal (Goal 1). Additional mitigative capability is examined as part of SBO (Guideline 3).

1.3.2 Control the Frequency of High-Consequence Sequences

Based upon the inherent capability of the Mark III design (as noted above), any pathways that might open, which would allow the fission products to bypass the pool are undesirable.

Because of the importance of the suppression pool as a mitigative feature, the vulnerability of a Mark III containment to any bypass pathways should be examined. Guideline 1, as shown in Table 1.1, is provided to ensure the low frequency of interfacing systems LOCA events based on insights from PRAs for other plant types.

Guideline 1 - Interfacing Systems LOCA

Interfacing systems LOCAs represent the possibility of high releases and therefore make it important to ensure that the frequency of these events is kept very low at all Mark III plants.

1.3.3 Reduce High Core-Damage Frequency Sequences

The most important contributors to CDF for Grand Gulf were found to be SBO and ATWS sequences. However, other accident sequences that were not important contributors to CDF in Grand Gulf could become dominant for other Mark III plants because of the unavailability of certain plant features that were available in Grand Gulf. Therefore, the guidelines are intended to cover a large spectrum of accident sequences.

Guideline 2 - Anticipated Transients Without Scram

Anticipated transients without scram (ATWS) have been found to be important contributors to risk for many BWRs. The NRC promulgated the ATWS Rule to reduce the frequency of ATWS events. The criteria developed for this guideline emphasize human reliability insights as to the importance of operator training and correct emergency procedures to ensure recovery from an ATWS event.

Guideline 3 - Station Blackout

For accidents involving the loss of offsite power and onsite emergency power, the NRC recommends examining the proposed station blackout (SBO) rule for applicability. The criteria associated with Guideline 3 are intended to emphasize the need to search for plant specific features and potential common cause failures which could disable systems required to work during an SBO. For individual plants which are found to have a vulnerability to SBO, the criteria highlight the importance of a proper emergency procedures and operator training in recovering from an SBO event. Note that this guideline is also related to Goal 1 (See Section 1.3.1).

Guideline 4 - Loss of Containment Heat Removal Sequences

Accident sequences involving loss of containment heat removal (CHR) were found to be quite important in the earlier PRA studies reviewed. In WASH-1400 loss of CHR sequences accounted for 53% of the calculated CDF. In the Grand Gulf RSSMAP study, these sequences similarly accounted for about 50% of the calculated CDF. However, the most recent Grand Gulf studies (IDCOR and ASEP/SARRP) show a two-and-three-order-of-magnitude reduction, respectively, in the quantification of these sequences. Therefore, Guideline 4 has been developed to generically address the mechanisms already effectively employed at Grand Gulf to reduce the frequency of loss of CHR sequences.

Guideline 5 - Support System Interdependencies

Although the importance is difficult to quantify, one of the insights developed in most risk assessment studies is the importance of support system interdependencies. For example, a preliminary draft of the ASEP Peach Bottom study indicated that loss of all service water was a dominant contributor to core damage. The final version of the accident sequence studies has reduced it to one percent of the overall CDF. In order to ensure that support system vulnerabilities do not cause unacceptably high CDFs for BWR Mark III plants, the authors have provided Guideline 5 to help assess any weaknesses of the support systems.

Guideline 6 - Maintain Containment Integrity

For sequences that threaten the containment by overpressure, containment venting has the potential to preserve the containment structural integrity by relieving noncondensable gases and/or saturated steam. Both the IDCOR and Severe Accident Risk Reduction Program (SARRP) results indicate that, provided drywell integrity is maintained, containment venting does not significantly reduce the fission products released to the environment. However, preserving the structural integrity of the containment is potentially important to ensuring operability of essential equipment. For accident sequences resulting in loss of CHR before core damage, venting may reduce the likelihood of core damage. In addition, the possibility of structural failure of the containment in the wetwell causing large fission-product releases has not been precluded for all Mark III plants. Thus, a guideline for containment venting has been developed.

Guideline 7 - Flooding Within Reactor Building

Flooding of the reactor building has been found to be a significant contributor to core damage in only one Mark II plant. However, the concerns appear to be of general applicability to other designs. Thus, this guideline was developed for Mark III plants to assess the potential for flooding of safety related equipment.

1.4 Using the Guidelines and Criteria

Numerous investigations, including PRAs, have been performed for the reference plants and similar plants by both the NRC and the nuclear power industry. The insights gained from many of the studies have been used in developing the guidelines and criteria contained in this report (including the other volumes of this report). These guidelines and criteria are issued to provide guidance to the analyst performing an IPE. This guidance is in the form of plant features, operator actions and the criteria for assessing those features and actions found to be helpful in reducing the overall risk for Grand Gulf and other Mark III plants. Thus, the guidance is given to provide a resource in examining the subject plant to determine if the same, or similar, guidelines will be of value in reducing overall plant risk. These guidelines and criteria are intended to be used solely as guidance.

1.5 References for Section 1

1. "Grand Gulf Nuclear Station - Integrated Containment Analysis," IDCOR Technical Report T23.1GG, March 1985.

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

3. C. N. Amos et al., "Containment Event Analysis for Postulated Severe Accidents: Grand Gulf Nuclear Station, Unit 1," Sandia National Laboratories, NUREG/CR-4700, Vol. 4, Draft for Comment, April 1987.

4. C. N. Amos et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Grand Gulf, Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 4, Draft for Comment, April 1987.

5. S. W. Hatch et al., "Reactor Safety Study Methodology Applications Program: Grand Gulf #1 BWR Power Plant," Sandia National Laboratories, NUREG/CR-1659/4 of 4, October 1981.

6. GESSAR (General Electric Standard Safety Analysis Report) II BWR/6 Nuclear Island, Probabilistic Risk Assessment.

Table 1.1 Guidelines for Preventing and Mitigating Severe Accidents in a BWR with a Mark III Containment

Guideline	Description
	<u>Control the Frequency of High-Consequence Sequences</u>:
1	Interfacing Systems LOCA
1.A.	Prevent Overpressurization of Low Pressure Systems
	<u>Reduce High Core-Damage Frequency Sequences</u>:
2	Anticipated Transients Without Scram (ATWS)
2.A.	Provide Operator Response During ATWS
3	Station Blackout
3.A.	Provide Reactor Pressure Vessel Injection
3.B.	Provide Hydrogen Control
4	Loss of Containment Heat Removal Sequences
4.A.	Provide Emergency Core Cooling Injection
5	Support System Interdependencies
5.A.	Examine Support System Interdependencies
6	Maintain Containment Integrity
6.A.	Provide Containment Venting Capability
7	Flooding Within Reactor Building
7.A.	Prevent and Mitigate Reactor Building Flooding

2. INTRODUCTION

2.1 Background

The U.S. Nuclear Regulatory Commission (NRC) has formulated an approach for a systematic safety examination of existing plants to determine whether particular severe accident vulnerabilities are present and what changes are desirable to ensure that there is no undue risk to public health and safety.

The Industry Degraded Core Rulemaking Program (IDCOR) selected four reference plants for detailed analysis, namely:

- Peach Bottom (a BWR with a Mark I containment)
- Grand Gulf (a BWR with a Mark III containment)
- Zion (a PWR with a large dry containment)
- Sequoyah (a PWR with an ice condenser containment)

The IDCOR analyses performed for the above reference plants have been documented together with the methodology used for the analyses and the technical basis supporting the methodology.

Parallel with the IDCOR work, the NRC under the Severe Accident Research Program (SARP), performed risk assessments, audit calculations, sensitivity studies, and uncertainty analyses for five plants. The five plants considered include the above four IDCOR reference plants, and, in addition

- Surry (a PWR with a subatmospheric containment)

The purpose of this effort is to review all of the IDCOR and SARP analyses performed for the reference plants, understand the reasons for the differences, and then use the experience gained from these reviews for developing guidelines and criteria that identify plant features and operator actions found to be important for either preventing or mitigating severe accidents in each plant type. In turn, these guidelines should be helpful in the systematic safety examination of individual plants.

The first plant reviewed was Peach Bottom, which is a BWR-4 with a Mark I containment. The IDCOR Peach Bottom analysis[1] was documented in March 1985 and was supplemented by additional sensitivity studies in July 1985. The SARP Peach Bottom reports[2-4] were reviewed in draft form during 1986. These reports were published early in 1987 and were summarized in the "Reactor Risk Reference Document" (NUREG-1150),[5] which was published for comment in February 1987. The experience gained from the review of these Peach Bottom studies along with other BWR PRA studies (namely, Limerick, Shoreham and Browns Ferry) was used to generate the guidelines and criteria which are the subject of Volume 1 of this report.

This volume builds on the experience gained during the Brookhaven National Laboratory (BNL) Peach Bottom review and deals with severe accidents in a BWR-6 with a Mark III containment (BWR Mark III). Both IDCOR and SARP used the Grand Gulf plant as the reference plant for this class of reactors so this report is based largely on analyses of severe accidents at Grand Gulf although other relevant PRAs were considered. The IDCOR Grand Gulf analysis[5] was documented in March 1985 whereas the SARP Grand Gulf reports[7-9] were

reviewed in draft form during 1986. The SARP reports were published in early 1987 and were summarized in NUREG-1150.[5]

2.2 Objectives

Three basic objectives or goals for this severe accident program apply equally to all plant types:

- Goal 1: Mitigate fission-product releases.
- Goal 2: Control the frequency of high-consequence sequences.
- Goal 3: Reduce high core-damage frequency.

The aim was, therefore, to develop detailed guidelines and criteria that could be used to achieve these goals during the examination of individual plants. The guidelines and criteria are defined in the following sections.

2.2.1 Guidelines

"Guidelines," as used in this report, identify those plant features and operator actions that were found to be important to either preventing or mitigating severe accidents in the reference plant studies. These guidelines provide a list of plant features and operator actions that the utilities can use as part of each individual plant examination (IPE). It is not the intent of this report to specify a set of improvements for either the reference plant or for any other plant which would be sufficient to achieve a certain level of safety. Instead, the guidelines indicate potential improvements in various areas of plant design and operation of which each utility should be aware when conducting its IPE and making decisions on plant improvements. The intent is to provide guidance to the analyst performing an IPE, as to the plant features and operator actions which were found to reduce overall risk. It is prudent to check whether potential improvements identified in studies of other similar plants can be of help in improving overall plant performance. The guidelines contained in this report, therefore, complement the IPEs.

The three objectives or goals were noted as applying equally to all plant types. Although the goals are independent of plant type, the guidelines that are needed to achieve the goals are plant dependent. In general terms, Goal 1 implies that there should be effective means of mitigating the fission-product releases for the broad classes of accident sequences which dominate the core-damage frequency. Therefore, these dominant accident sequences have to be determined and those plant features and operator actions that are available to mitigate the release of fission products have to be identified. Only then can detailed guidelines be developed to ensure that these dominant accident sequences can be mitigated.

There may be accident sequences for which a specific plant will have substantial fission-product releases (e.g., containment bypass sequences). Thus, for such sequences Goal 1 may be difficult to achieve. Therefore, all reasonable steps are identified which could reduce the frequency of these potentially high-consequence sequences (namely Goal 2). Again, the accident sequences have to be identified and plant vulnerabilities and/or operator actions that lead to core damage for these sequences also have to be identified. Detailed guidelines can then be developed which will aid in

assessing an individual plant's capability to prevent these sequences from occurring.

It is also desirable to ensure that the overall core-damage frequency is low (namely Goal 3). Again, the dominant accident sequences have to be found so that detailed guidelines can be developed to reduce the frequency of these sequences, if necessary.

In general, the following screening process was used to determine whether or not to develop a particular guideline:

- any accident sequence with a core-damage frequency greater than 10^{-6} per reactor year
- any sequence that contributed to more than 5% of the total core-damage frequency
- any event that caused a conditional probability of early containment failure greater than 0.1
- any sequence that resulted in containment bypass with a frequency greater than 10^{-7} per reactor year
- any sequence that was judged to be uniquely important (example, very severe consequences)

This screening process has led to the development of guidelines that can be used in the systematic safety examination of other BWRs with Mark III containments. For example, the guideline for venting of the wetwell was identified as an item that would help to achieve Goal 3 (namely, reduce high core-damage frequency) for the BWR Mark III reference plant. Therefore, in the safety examination of other BWRs with Mark III containments, the need for containment venting may need to be carefully assessed.

The development of a particular guideline for the BWR Mark III reference plant does not imply that this plant or any of the other plants in this category need to conform to this guideline. It simply means that analyses have indicated that this particular guideline has the potential to significantly reduce risk. Thus, the guidance is given to provide a resource in examining the subject plant to determine whether the same or similar guidelines will be of value in the reduction of overall plant risk. Whether or not the guideline is useful or needed in a particular BWR with a Mark III containment depends on plant-specific details and is beyond the scope of this report and is therefore not addressed here.

2.2.2 Criteria

"Criteria," as used in this report, are the attributes which have been identified as important to assess the performance of plant features and operator actions identified in the guidelines. The criteria provide deterministic (as opposed to probabilistic) performance measures which are judged to be helpful to implement the concepts contained in the guidelines. When a decision is made to provide an item or an action specified in a guideline, the utility should address a set of questions relating to the design, operation

and availability of the needed equipment and the training of operators. For the example of containment-venting guidance, the capacity of the venting system, the selection of setpoints to initiate venting, the availability of applicable procedures and the accessibility of certain valves by operators should be addressed. The criteria on containment venting provide helpful information in assessing venting capability in each individual plant.

The criteria address the general issues of (1) survivability of equipment (i.e., whenever credit is given for a system or a component to mitigate the accident, the ability of the equipment to function under environmental and fluid dynamic loads associated with severe accident sequences must be taken into account), (2) equipment capabilities, capacities, and duration of operability, (3) accessibility of equipment, (4) availability of support systems, (5) identification of necessary components, (6) identification of important operator actions, and (7) identification of parameters for initiation of mitigating systems and operator actions.

2.3 Organization of the Report

This report describes detailed guidelines and criteria for preventing and mitigating severe accidents in BWRs that have a Mark III containment. It is the third volume in a series of five, that deal with guidelines and criteria for several different reactor and containment types. Other volumes in the series are:

- Volume 1: BWRs with Mark I Containments
- Volume 2: BWRs with Mark II Containments
- Volume 4: PWRs with Large Volume Containments
- Volume 5: PWRs with Ice Condenser Containments.

Appendix A of this volume contains a review of the IDCOR and SARP analyses for a BWR-6 with a Mark III containment along with other pertinent studies. The insights gained from these studies lead to the identification of the strengths and vulnerabilities of a BWR with a Mark III containment. In Section 3, the three basic goals of the program are related to the relevant design features and operating characteristics of a BWR-6 with a Mark III containment. The guidelines recommended to achieve the three goals are therefore initially developed in Section 3. In Section 4, the guidelines are restated and detailed criteria are developed for each guideline.

2.4 References for Section 2

1. "Peach Bottom Atomic Power Station-Integrated Containment Analyses," IDCOR Technical Report T23.1PB, March 1985.

2. A. M. Kolaczkowski et al., "Analysis of Core Damage Frequency from Internal Events: Peach Bottom, Unit 2," Sandia National Laboratories, NUREG/CR-4550, Volume 4, October 1986.

3. C. N. Amos et al., "Containment Event Analysis for Postulated Severe Accidents: Peach Bottom Atomic Power Station, Unit 2," Sandia National Laboratories, NUREG/CR-4700, Vol. 3, Draft Report for Comment, May 1987.

4. C. N. Amos et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Peach Bottom, Unit 2," Sandia National Laboratories, NUREG/CR-4551, Vol. 3, Draft for Comment, April 1987.

5. "Reactor Risk Reference Document," U.S. Nuclear Regulatory Commission, NUREG-1150, Draft for Comment, February 1987.

6. "Grand Gulf Nuclear Station - Integrated Containment Analysis," IDCOR Technical Report T23.1GG, March 1985.

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

8. C. N. Amos et al., "Containment Event Analysis for Postulated Severe Accidents: Grand Gulf Nuclear Station, Unit 1," Sandia National Laboratories, NUREG/CR-4700, Vol. 4, Draft for Comment, April 1987.

9. C. N. Amos et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Grand Gulf, Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 4, Draft for Comment, April 1987.

3. DEFINITION OF GOALS AND RELEVANT BWR MARK III FEATURES

In Section 2 of this report, the concept of three basic objectives or goals for this severe accident program was introduced. The concept applies equally to all plant types. In this section, the three goals are related to the relevant design features and operating characteristics of a BWR-6 with a Mark III containment for the accident sequences and containment failure modes found to be important in Appendix A. This includes consideration of both favorable and unfavorable severe accident attributes.

Screening criteria have been used to identify those sequences which need to be addressed by severe accident guidelines for each goal. Specifically:

- For Goal 1 (Mitigate fission-product releases), all sequences have been examined which represent at least 5% of the core-melt frequency or are estimated to occur more often than 10^{-6} per reactor year and which result in a conditional probability of early containment failure greater than 0.1.
- For Goal 2 (Control the frequency of high-consequence sequences), all sequences have been examined which result in pool bypass and are estimated to occur more often than 10^{-7} per reactor year.
- For Goal 3 (Reduce high core-damage frequency sequences), all sequences have been examined which "have the potential to occur" more frequently than 10^{-6} per reactor year. Note that this screening criterion has been used to identify potential vulnerabilities from risk assessment insights which do not necessarily apply to Grand Gulf itself, but may apply to other Mark III plants.

This section provides the link between the goals (developed in Section 2) and the guidelines (developed in Section 4) that may be used to assess the capability of specific plants to meet these goals. This section is organized into three subsections, which correspond to the three goals.

3.1 Mitigate Fission-Product Releases

This goal requires that there shall be effective means of mitigating the fission-product releases for the broad classes of accident sequences that may lead to core damage in a BWR with a Mark III containment. In Appendix A, the most important contributors to the core-damage frequency (CDF) were found to be SBO and ATWS sequences. Other transients and loss-of-coolant accidents (LOCAs) may also contribute to the CDF. Accident sequences for which mitigation by the Mark III containment are ineffective are also identified in Appendix A. These specific sequences are discussed in Section 3.2, which attempts to determine how the frequency of these unmitigated sequences can be reduced. This section concentrates on the broad classes of accident sequences for which plant features provide significant means of mitigating fission-product release. In the following sections both the favorable and unfavorable severe accident attributes of the Mark III containment are identified.

3.1.1 Plant Vulnerabilities

The Mark III containment is a pressure-suppression design. The suppression pool is available to condense steam released from the primary system

during an accident. However, the Mark III containment may be vulnerable to pressure and/or temperature buildup because of the noncondensable gases and heat released during a core-meltdown accident. There are differences between the IDCOR[1] and Battelle Columbus Laboratories (BCL)[2] (NUREG-1150) analyses regarding the estimates of how long it will take to pressurize a Mark III containment to its ultimate capacity after the core debris has failed the reactor vessel (and is interacting with concrete) but both studies concluded that the containment pressure boundary will eventually fail. Therefore, unless mitigative actions are taken, a Mark III containment will fail eventually because of overpressure or overtemperature. If drywell leakage occurs, a fraction of the fission products in the drywell atmosphere could pass to the secondary outer containment pressure boundary (and ultimately to the environment) without the benefit of suppression pool scrubbing. Note that suppression pool scrubbing is an important mitigative feature of a Mark III containment (refer to Section 3.1.2).

An inspection of the Mark III containment configuration in Appendix A (see Figure A.2) will show that the pedestal below the reactor pressure vessel would tend to confine the core debris after a core meltdown accident. In the absence of a water supply (no water from a LOCA, upper pool dump or restoration of reactor coolant injection), extensive core-concrete interactions would be expected to occur. There are differences between the IDCOR and SARP analyses as to how hot the core debris will remain during these interactions and as to how many of the less volatile fission products will be released. IDCOR assumes that water from the lower plenum and control rod drive (CRD) cooling pumps will be available to quench the core debris and keep it cool as long as CRD flow is available. However, at this time, the authors do not believe that the possibility of the core debris remaining hot and releasing significant quantities of fission products has been ruled out particularly under SBO conditions.

After the reactor vessel fails there would be sufficient core materials in a full core-meltdown to fill the cavity. If the core debris remained molten it could erode the pedestal support and cause the vessel to be displaced resulting in failure of the drywell wall. This is a mechanism for early loss of drywell integrity and is thus another Mark III containment vulnerability.

Both IDCOR[1] and BCL[2] predict that a substantial quantity of hydrogen will be produced during core degradation and core-concrete interaction. The hydrogen provides both a temperature and a pressure threat to containment. If the hydrogen burns, the high temperatures and pressures provide a threat to drywell integrity which may lead to pool bypass (as modeled in the SARP analysis). The Grand Gulf containment is equipped with hydrogen igniters which are intended to ensure that the hydrogen does not accumulate to explosive concentrations. However, the igniters depend on ac power. Therefore, they would not be available during blackout sequences. Even if the igniters perform their intended function, the resulting high temperatures may contribute to drywell penetration failure.

In the sections that follow, suppression pool scrubbing is noted as an effective mitigative feature for the Mark III containment provided all of the fission products pass through the pool. It is, therefore, important to ensure that paths do not open which would allow the fission products to bypas the

suppression pool. The vacuum breakers between the wetwell and drywell would create a path that bypasses the suppression pool, if they fail open. In addition, the various drywell penetration seals could be degraded at high temperatures and pressures. Failure of these seals would also open up paths which could bypass the suppression pool. If the main steam isolation valves (MSIVs) fail to close, another suppression pool bypass path would exist.

Although severe accidents in the Mark III containment may cause high temperatures, high pressures and hydrogen combustion, the containments have several very important mitigative features, which are described in the section that follows.

3.1.2 Mitigating Features

The suppression pool in a Mark III containment is a very effective mechanism for trapping any fission-product aerosols that might pass through it. Thus, to a large extent, the suppression pool has the potential to compensate for the vulnerabilities identified above (in Section 3.1.1). For example, overpressure failure of the containment can be prevented by venting. With venting containment integrity is lost but the containment function (retention of the fission products in the pool) is maintained.

High wetwell temperatures and possible seal degradation of penetrations through the drywell wall can be prevented by containment spray. Containment spray will also contribute to decontamination even for sequences with substantial pool bypass.

The Mark III containment has hydrogen igniters which prevent hydrogen accumulation. This is a very significant mitigative feature, which is important to maintain during a severe accident. However, the igniters, as currently powered, would not be available during an SBO.

The above discussion has identified several plant features of the BWR plant with a Mark III containment that have the potential to help achieve Goal 1, namely, mitigating fission-product releases. Moreover, both IDCOR and SARP indicate (see Appendix A) that significant bypass (beyond design leakage) of the pool is very unlikely. With a low probability (<10%) of early pool bypass, additional mitigative guidelines and criteria do not appear to be justifiable and therefore the authors have not developed any guidelines in Section 4 to meet Goal 1. A relatively low likelihood of pool bypass is also indicated in the GESSAR PRA[3] and the Safety Evaluation[4] of GESSAR.

3.2 Control the Frequency of High-Consequence Sequences

The plant features identified in Section 3.1 (the suppression pool, containment sprays and hydrogen igniters), have been found to effectively mitigate fission-product releases for the broad classes of accident sequences that were found to dominate the core-damage frequency (CDF). However, accident sequences were identified in Appendix A for which the BWR-6, Mark III plant may not be effective in mitigating fission-product release.

3.2.1 Interfacing Systems LOCA

The interfacing systems LOCA would open up a path from the primary system, bypassing the primary containment and suppression pool completely. The only plant feature pertinent to mitigating this sequence is the auxiliary building, which is not sufficient on its own to ensure low fission-product release to the environment. The frequency of these potentially high-consequence accident sequences must, therefore, be maintained at acceptably low levels (Goal 2). Neither IDCOR nor Sandia National Laboratories (SNL) have identified the interfacing systems LOCA as a significant contributor to core melt frequency. However, since the consequences of an interfacing systems LOCA are potentially high and it is the subject of ongoing research (GI-105), a guideline has been developed and should be considered appropriate pending resolution of the issue. This guideline and associated criteria related to Goal 2 dealing with prevention of high consequence sequences are developed in Section 4.

3.3 Reduce High Core-Damage Frequency Sequences

In Appendix A, only a few accident sequences were found which figure prominently in the core-damage profiles of all of the PRAs reviewed. This led to the conclusion that if the frequency of this relatively small subset of accident sequences could be reduced, then the overall CDF could also be reduced. The most important contributors to the CDF were found to be SBO and ATWS sequences. Therefore, severe accident guidelines with specific detailed criteria have been developed in Section 4 related to these accident sequences. In the following sections, these "dominant" core-damage sequences are identified and discussed.

3.3.1 Station Blackout

Station blackout (SBO) refers to a loss of the offsite power system with concurrent failure of the two emergency ac power divisions. SBO is currently the subject of an unresolved safety issue (namely USI A-44). Reducing SBO sequences is also addressed by the proposed NRC SBO rule. The guidelines and associated criteria developed by the present study emphasize the need to search for plant specific features and potential common cause failures which could disable systems required to work during an SBO. For individual plants which are found to have a vulnerability to SBO, the criteria highlight the importance of proper emergency procedures and operator training in recovering from an SBO event.

For the BWR Mark III design, the two systems designed to operate in the presence of an SBO are the high-pressure core spray (HPCS) and the reactor core isolation cooling (RCIC) systems. By removing the long-term blackout sequence related to dependent failure modes of either system, the blackout CDF can be significantly reduced.

3.3.2 Anticipated Transients Without Scram (ATWS)

ATWS has been identified as a potentially significant contributor to the core-damage frequency in Appendix A. Therefore, a severe accident guideline has been developed in Section 4 related to these sequences. An ATWS rule has

been issued and compliance with this rule was assumed in the formulation of the detailed criteria for this guideline.

The guideline for ATWS and the detailed accompanying criteria do not address specific hardware/systems modifications as was proposed for the SBO guideline. This is based upon the observations in Appendix A that a fairly large number of improvements to hardware/systems have already been developed and implemented in the BWR Mark III design. Plants that have or plan to incorporate these design features will have acceptably reduced the ATWS core damage frequency without further hardware or systems modifications. It therefore must be stressed that the ATWS guideline and associated criteria in Section 4 assume that incorporation of the design features noted in Appendix A have or will be incorporated into the design (alternative rod insertion (ARI) and high flow "equivalent" standby liquid control (SLCS) system).

3.3.3 Loss of Containment Heat Removal Sequences

Accidents involving loss of containment heat removal (CHR) were found to be important in the Grand Gulf RSSMAP report.[5] These accidents were found not to be important in the IDCOR and ASEP analyses for Grand Gulf because of credit given in these studies for containment venting (or other containment leakage) and alternative injection capability. Therefore, based upon engineering judgement, it has been deemed prudent to establish a guideline on this subject with associated criteria. The underlying purpose of this guideline is to ensure that other Mark III plants will have the features and capabilities that validate the assumptions and credit given in the IDCOR and SARP analyses. Guidelines and detailed criteria to ensure that loss of CHR sequences do not lead to core damage are developed in Section 4.

3.3.4 Support System Interdependencies

Most PRAs have stressed the importance of unrecognized interdependencies having the potential to compromise the performance of many critical safety systems. In many cases, risk assessment studies have identified such vulnerabilities very early in the study and "fixes" have been made which substantially reduced risk. Although no such dependency-caused vulnerability has been identified for Grand Gulf, "engineering judgement" indicates that it may be useful to search for the existence of such interdependencies in other Mark III plants.

3.3.5 Flooding Within the Reactor Building

One of the accident sequences, whose potential for contributing to the core damage frequency was specifically evaluated in the Shoreham Nuclear Power Station (SNPS) PRA,[6] is the release of excessive water into the reactor building. Both the SNPS PRA and the BNL review[7] of the SNPS PRA revealed that accident sequences induced by such an initiator contribute substantially to the core damage frequency (3.9×10^{-6} and 2.0×10^{-5}/reactor year, respectively).

To help ensure that Mark III plants which may have a similar safety-related equipment flooding potential can be identified, a guideline and associated criteria are provided in Section 4.3 which may be used to screen for such vulnerabilities.

3.4 References for Section 3

1. Grand Gulf Nuclear Station, "IDCOR Task 23.1 Integrated Containment Analysis," October 1984.

2. R. S. Denning et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios: BWR, Mark III Design," Battelle Columbus Laboratories, NUREG/CR-4624, Vol. 4, July 1986.

3. GESSAR (General Electric Standard Safety Analysis Report) II BWR/6 Nuclear Island, Probabilistic Risk Assessment.

4. GESSAR II Safety Evaluation Report, USNRC, NUREG-0979, April 1983.

5. S. W. Hatch et al., "Reactor Safety Study Methodology Applications Program: Grand Gulf #1 BWR Power Plant," Sandia National Laboratories, NUREG/CR-1659/4 of 4, October 1981.

6. "Probabilistic Risk Assessment - Shoreham Nuclear Power Station," Long Island Lighting Company, June 1983.

7. D. Ilberg et al., "A Review of the Shoreham Nuclear Power Station Probabilistic Risk Assessment (Internal Events and Core Damage Frequency)," Brookhaven National Laboratory, NUREG/CR-4050, June 1985.

4. GUIDELINES AND CRITERIA FOR A BWR WITH A MARK III CONTAINMENT

In Section 3 those accident sequences that dominate the core-damage frequency were identified as were those that are potentially of high consequence. Vulnerabilities of the Mark III containment to severe accident containment loads were discussed and those features of a BWR with a Mark III containment, which are important for preventing core damage and are available for mitigating fission-product release to the environment were identified.

Based on the "insights" from previous PRAs and other severe accident research, the following sections provide guidelines defining "deterministic, plant-specific guidance on the design features and operating characteristics which are to be examined by the utilities,"[1] and criteria defining "deterministic standards for judging the acceptability of plant features."[1] From SECY-86-76,[2] further guidance is provided in defining guidelines and criteria. These guidelines "will specify the plant features and operator actions which are considered important to ensuring acceptable risk for the reference plant."[2] Further acceptance criteria (for the various guidelines) "will specify the attributes necessary to ensure acceptable performance."[2]

Based on this work, seven guidelines were developed which reflect the importance of these features to plant risk. As discussed in Section 2.2.1, these guidelines indicate areas of potential improvements for various areas of plant design and operation of which utilities should be aware when conducting individual plant assessments. It is further noted that a number of the guidelines appear to overlap various generic issues as defined by the NRC. Final resolution and disposition of these generic issues may encompass NRC-imposed requirements. However, the guidelines and criteria presented herein are intended only for the purposes noted above. The guidelines are summarized in Table 1.1.

No guidelines were identified as being justified to be developed to ensure the capability to mitigate fission-product releases (Goal 1) since current research indicates that the Mark III containment will provide sufficient mitigation.

Guideline 1 was developed for controlling the frequency of high-consequence sequences (Goal 2) with reference to minimizing interfacing systems LOCA frequency.

Guidelines 2 through 7 were developed for reducing high overall core-damage frequency sequences (Goal 3) with reference to mitigating anticipated transients without scram (ATWS) sequences, mitigating station blackout (SBO) sequences, mitigating loss of containment heat removal (CHR) sequences, examining support system interdependencies, providing containment venting capability, and mitigating floods within the reactor building.

The remainder of this section is organized into three subsections corresponding to the three basic goals. In each subsection, the corresponding guidelines are discussed from which detailed criteria are developed in order to provide the standards by which each plant could be measured for compliance with the guidelines. The criteria address (see Section 2.2.2) the general issues of (1) operability and survivability of equipment and systems (i.e., whenever credit is given for a system or component to mitigate the accident,

the ability of the equipment to function under the environmental conditions and fluid dynamic loads associated with severe accident sequences must be taken into account), (2) capability and capacity of equipment, (3) reliability and accessibility of equipment, (4) availability of support systems, (5) identification of necessary components and operator actions, and (6) parameters for initiation of mitigating systems and operator actions.

4.1 Mitigate Fission-Product Releases

The review of the performance of a Mark III containment for the dominant core damage sequences indicated that pool bypass is unlikely so that no guidelines were required to ensure the capability to mitigate fission-product releases.

4.2 Control the Frequency of High-Consequence Sequences

Accident sequences were found in Appendix A for which the Mark III containment has limited means of mitigating fission-product releases, namely an interfacing systems LOCA. In this section a guideline and associated criteria for controlling these potentially high-consequence sequences are developed. Both IDCOR and SARP estimate these sequences to be small contributors to core melt for Grand Gulf. This guideline has been provided to ensure that other Mark III plants keep these high-consequence sequences at a low level.

4.2.1 Interfacing Systems LOCA (Guideline 1)

In general BWR Mark III PRAs have found the interfacing systems LOCA (ISL) to be a highly unlikely event (less than 10^{-7}/reactor year). However, there are some BWRs (e.g., Shoreham) for which the ISL is risk significant because of the potentially high releases. The objective of this guideline and associated criteria is to ensure that the frequency of ISL events is kept at an acceptably low level. Brookhaven National Laboratory (BNL) is presently performing a study to provide technical support to the NRC for the meaningful resolution of the generic issue related to ISL (GI-105). Therefore, the criteria for this guideline should be considered appropriate pending resolution of the generic issue.

In order to control the frequency of ISL sequences, specific performance criteria have been developed to assess the performance of equipment, systems and operators. The criteria relate to equipment (low-pressure systems interfacing with high-pressure systems) and operator performance (isolation and relief valve maintenance and surveillance).

Detailed criteria developed for this guideline are given in Table 4.1.

4.3 Reduce High Core-Damage Frequency Sequences

The major contributors to the core-damage frequency (CDF) are presented in Section 3.3. The IDCOR and ASEP/SARP analyses imply that the SBO and ATWS sequences are the dominant contributors to the CDF. The results of other PRAs and PRA reviews indicate that in addition to those two types of sequences, other sequences, namely, loss of containment heat removal (CHR) sequences (TW, SI, TQUV, and TPQI sequences) and sequences with failure to depressurize the

reactor pressure vessel (RPV) for injection with low-pressure systems (TQUX sequences), can also be major contributors to the CDF.

4.3.1 Anticipated Transients Without Scram (ATWS) (Guideline 2)

The important attributes of this sequence with respect to operator actions were found[3] to be the likelihood of misleading instrumentation, the need to inhibit automatically initiated safety systems, the use of required mitigating actions which conflict with operator response to other accident conditions, and the need for coordinated actions and communication among control room crew members under highly stressful conditions.

For the ATWS guideline, the performance of equipment, systems, and operators should be assessed against specific performance criteria to ensure successful use of this guideline. The criteria relate to the equipment, systems, and operator performance by emphasizing operator familiarization, aids, and understanding of potentially conflicting signals.

Detailed criteria developed for this guideline are given in Table 4.2 and are based upon the assumption that each of the plants is (or will be) in compliance with the NRC rule on "Reduction of Risk from Anticipated Transients Without Scram for Light-Water-Cooled Nuclear Power Plants."[4]

4.3.2 Station Blackout (Guideline 3)

In most PRAs for light-water-reactors (LWRs), station blackout (SBO) sequences have been prominent contributors to the CDF. As part of the effort to resolve the unresolved safety issue (USI A-44), the NRC is proposing to amend its regulations "to provide further assurance that an SBO (loss of both offsite power and onsite emergency ac power systems) will not adversely effect the public health and safety."[5] For accident sequences, developed by an individual plant examination (IPE), which involve the loss of offsite power and onsite emergency power, the proposed SBO rule should be examined for applicability. The criteria associated with Guideline 5 are intended to emphasize the need to search for plant specific features and potential common cause failures which could disable systems required to work during an SBO. For individual plants which are found to have a vulnerability to SBO, the criteria highlight the importance of proper emergency procedures and operator training in recovering from an SBO event.

The performance of equipment, systems and operators should be assessed against specific performance criteria to ensure successful accomplishment of this guideline. The criteria relate to the equipment, systems, and operator performance as follows:

- equipment needs with respect to cooling,
- equipment needs for dc power, and
- operator understanding of the above equipment needs and their limitations.

The criteria developed for this guideline are given in Table 4.3.

4.3.3 Loss of Containment Heat Removal (CHR) Sequences (Guideline 4)

In the Grand Gulf RSSMAP[6] report, sequences with successful coolant injection but with subsequent loss of containment heat removal (e.g., SI and TPQI sequences in Table A.1) were important contributors to the CDF. In that study it was assumed that containment failure caused loss of injection. As discussed in Appendix A, in the PRAs where those sequences are not important, the main factor for the low contribution to CDF is the credit given for containment venting and alternative sources of injection. It would therefore appear that alternative injection sources should be available in addition to containment venting to provide adequate CHR during accident sequences with successful coolant injection but with subsequent loss of CHR.

The performance of equipment, systems, and operators should be assessed against specific performance criteria to ensure successful prevention of core damage for loss of CHR sequences. The criteria relate to the equipment, systems, and operator performance as follows:

- source of cooling water,
- means of supplying the water,
- instrumentation and controls to monitor and direct the water, and
- operator aids, familiarization and expertise to initiate, control, and terminate the water.

Previous studies have shown that injection is not lost in TW sequences for 20 or more hours. Thus, alternative injection injection sources should be capable of removing decay heat at that time (about 0.5%).

Detailed criteria developed for this guideline are given in Table 4.4.

4.3.4 Support System Interdependencies (Guideline 5)

One of the primary benefits of performing a rigorous PRA is that the system interdependencies are modeled and are reflected in the results. However, not all PRA studies have performed rigorous interdependence analyses and, therefore may not have ferreted out all of the possible subtle interdependencies. This may have profound effects upon their results. An interdependency is defined as the failure of one system leading directly or indirectly to the failure of another system.

An in-depth application of basic PRA methodology with respect to interdepencies yielded significant findings on a previously heavily studied PWR. To illustrate this point, reference is made to the BNL study of system interactions (support system interdependencies) at Indian Point Unit 3.[7] The major finding of that study was that a specific single station emergency battery could fail and among other things, negate the entire low-pressure injection function. The point to be emphasized here is that none of the numerous other studies and reviews of the Indian Point 3 design were able to detect this important single failure nor did the BNL study until all the support systems were explicitly modeled, linked together (the fault tree linking approach[8]) and solved using the SETS computer code.[9]

NUREG-1150[10] has provided a thorough application of the latest PRA methods to five reference plants and the results point out numerous insights

into the importance of specific design differences among the studied plants. However, the NUREG-1150 authors emphasize the importance of support system differences and the difficulty of extrapolating the result from one plant to another.

It is not sufficient to make a single overall interdependency table of the front-line and support systems for a given plant and simply compare that to the reference plant. No two plants will have the same set of system interdependencies. Support systems vary widely from plant to plant even though the plants may be of a similar class and have the same set of front-line systems.

It is recognized that following the steps outlined in Table 4.5, in a rigorous fashion, is a major undertaking. This fact, however, does not diminish its importance.

Based upon the dominance of the SBO sequences to the BWR designs, it is also recommended that a detailed interdependency table be constructed for this sequence with all interdependencies conditioned upon the existence of an SBO for various lengths of time. This table should also explicitly identify all of the expected failure mechanisms (e.g., identify whether battery failure is because of loss of room cooling or depletion).

4.3.5 Maintain Containment Integrity (Guideline 6)

For sequences that threaten the containment by overpressure, containment venting has the potential to preserve the containment structural integrity by relieving noncondensable gases and/or saturated steam. Both IDCOR and SARRP results indicate that venting is not important to the release fraction. However, preserving the structural integrity of the containment is potentially important to ensuring operability of essential equipment. For accident sequences resulting in loss of CHR prior to core damage, venting may reduce the likelihood of core damage. In addition, the possibility of failing the containment in the wetwell and allowing large fission product releases has not been precluded for all Mark III plants. Thus, it is recommended that emergency procedures for containment venting be implemented.

For the small subset of ATWS sequences with uncontrolled low pressure injection, the resultant high power level appears to preclude venting and the containment spray would be isolated. The criteria necessary to control the ATWS core-damage frequency are discussed in Section 4.3.1.

For the two dominant sequences of SBO and ATWS, venting procedures will be difficult to perform. For SBO sequences, power dependencies may preclude actuation of venting from the control room, and high radiation levels may hamper local manual actuation. For ATWS sequences, the large venting capacity requirements, short time frame for operator action and possible problems with normal isolation systems make successful venting under such conditions operationally difficult.

The relevant equipment, system, and human performance should be assessed by appropriate criteria to ensure successful accomplishment of containment venting as required during severe accident conditions. The criteria relate to the venting equipment, systems, and human performance as follows:

- means to vent the wetwell,
- instrumentation and controls to monitor and direct venting, and
- operator familiarization and expertise to initiate, control, and terminate the venting.

Detailed criteria developed for this guideline are given in Table 4.6.

4.3.6 Flooding Within the Reactor Building (Guideline 7)

Although medium or small leakages can be adequately mitigated by the existing sumps or pumpback systems, large water leakages are of primary concern in reactor building (RB) flooding. Potential water sources for excessive water release into the lowest RB level include the suppression pool, the condensate storage tank, the reactor coolant system, the service water system and the fire protection system storage tank. Some of the major equipment located in the lowest RB level compartment may include emergency core cooling (ECC) pumps and their electrical control panels for the high-pressure coolant injection (HPCI), RCIC, core spray, and low-pressure coolant injection (LPCI) systems.

RB flooding can be initiated by (1) a major maintenance which requires exposing a safety system to the RB atmosphere and (2) breaks in the pressurized or the non-pressurized part of piping or components. In item 1, "major maintenance" refers to those actions which would require dismantling of system components thus eliminating a barrier between large sources of water and the RB. RB flooding can partly be prevented and/or mitigated through proper training and procedures. For example, once the RB is flooded, the operator should be able to follow the instructions for responding to the alarm to identify the source of the flood and isolate it before the water level in the lowest compartment reaches a critical level. The operator should also know about alternative devices or equipment which can be utilized to provide coolant injection to the RPV in case of emergency core cooling (ECCS) systems equipment failures in the flooded compartment.

The BNL study[11] of the Shoreham Nuclear Power Station (SNPS) revealed that although the SNPS Alarm Response Procedures specify general guidelines for monitoring system parameters to determine the leakage location and initiate the leakage isolation, specific requirements for operators to systematically check the operation parameters of relevant systems are not included. BNL also identified that the random failure of an equipment protection electric circuit breaker coinciding with RB flooding may result in the propagation of failures to the upstream motor control center (MCC), other MCCs, and the associated load centers. It is important that this or similar potential common-mode failures be avoided.

Although this type of vulnerability to flooding was identified for a Mark II plant, it is believed that the concerns are of general applicability to other designs. Thus, it is recommended that Mark III plants also be screened for flooding vulnerabilities.

Detailed criteria developed for this guideline are given in Table 4.7.

4.4 Using the Guidelines and Criteria

Numerous investigations, including PRAs, have been performed for the reference plants and for similar plants by both the NRC and the nuclear power industry. The insights gained from many of the studies have been used in developing the guidelines and criteria contained in this report (including the other volumes relating to other plant types). The guidelines and criteria are issued to guide the analyst performing an IPE. This guidance is in the form of plant features, operator actions and the criteria for assessing those features and actions found to be helpful in reducing the overall risk for Grand Gulf and other Mark III plants. Thus, the guidance is given to provide a resource in examining the subject plant to determine if the same, or similar, guidelines will be of value in reducing overall plant risk. These guidelines and criteria are intended to be used solely as guidance, but they may include (as a subset) some requirements generated by the NRC on generic issues.

4.5 References for Section 4

1. R. Barrett, "Status of the Severe Accident Program for Operating Reactors," NRR Staff Presentation to the ACRS Subcommittee Class 9 Accidents, February 24, 1986.

2. SECY-86-76," Implementation Plan for the Severe Accident Policy Statement and the Regulatory Use of New Source-Term Information," NRC/EDO, February 28, 1986.

3. W. Luckas et al., "A Human Reliability Analysis for the ATWS Accident Sequence With MSIV Closure at the Peach Bottom Atomic Power Station," Technical Report A-3272 4/86, Brookhaven National Laboratory, April 1986.

4. ATWS Final Rule - Code of Federal Regulations, Title 10, Section 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram Events for Light-Water Cooled Nuclear Power Plants," June 1984.

5. NRC Station Blackout Proposed Rule, Federal Register, Volume 51, No. 55/March 21, 1986, pgs. 9829-9835.

6. S. W. Hatch et al., "Reactor Safety Study Methodology Applications Program: Grand Gulf #1 BWR Power Plant," Sandia National Laboratories, NUREG/CR-1659/4 of 4, October 1981.

7. R. Youngblood et al., "Fault Tree Application to the Study of Systems Interactions at Indian Point 3," NUREG/CR-4207, January 1986.

8. American Nuclear Society and Institute of Electrical and Electronics Engineers, "A PRA Procedures Guide," NUREG/CR-2300, January 1983.

9. R. B. Worrell and D. W. Stack, "A SETS User's Manual for the Fault Tree Analyst," Sandia National Laboratories, NUREG/CR-0465, SAND77-2051, November 1978.

10. "Reactor Risk Reference Document," U.S. Nuclear Regulatory Commission, NUREG-1150, Draft for Comment, February 1987.

11. K. Shiu et al., "A Review of the Accident Sequences Following an Excessive Release of Water at Elevation 8' of the Reactor Building in the Shoreham Nuclear Power Station," Draft Report, NUREG/CR-4049, BNL-NUREG-51835, March 1985.

Table 4.1 Criteria for BWR Mark III Containment
Guideline 1: Interfacing Systems LOCA

Concern: Although the interfacing system LOCA sequences have not shown themselves to be leading contributors to core-damage frequency, they represent potentially high release sequences and they appear to contribute significantly to the overall risk.

Function: Maintain Primary System Integrity (Guideline 1.A)

Guideline 1.A. Prevent Overpressurization of Low-Pressure Systems

Basis: Implementation of the following criteria will ensure that the frequency of an interfacing systems LOCA will remain acceptably low.

Criteria:

Note: Resolution of the Generic Issue 105 (GI-105), which deals with interfacing systems LOCAs for both BWRs and PWRs, may have an impact on this guideline. Therefore, the criteria below should be considered as appropriate pending resolution of GI-105.

1.A.1. All low pressure lines that potentially could be overpressurized should be identified and should be provided with alarms to alert the operator to the symptoms of an overpressure event.

1.A.2. The equipment designated to provide isolation and prevent overpressurization, such as the RHR line isolation valves or the low-pressure injection system check valves, should periodically undergo operability testing and local leak rate testing (LLRT).

1.A.3. The relief capability of the relief valves designated to mitigate low pressure system overpressurization should be established. In most if not all cases, these relief valves were not sized with the possibility of an interfacing systems LOCA in mind. However, given that an interfacing systems LOCA occurs in a non-isolatable portion of a low-pressure system, there may be alternatives available to the operator such as taking advantage of additional relief valves. If such or similar actions are found to be helpful, they should be factored into the appropriate emergency procedures.

1.A.4. Operator training and procedures should specify the actions to be taken to isolate the low pressure systems identified above or to depressurize the primary system, thereby mitigating the consequences of the interfacing systems LOCA.

1.A.5. After each reactor shutdown and cooldown, the isolation function of the pressure isolation valves should be tested. These valves should not be performed under reactor operating conditions.

Table 4.2 Criteria for BWR Mark III Containment Guideline 2: Anticipated Transients Without Scram (ATWS)

Concern: ATWS sequences have been shown to be one of the leading classes of severe-accident sequences in terms of core-damage frequency for most BWRs. Although Grand Gulf was found to have a low ATWS frequency (see Appendix A) other Mark III plants may not have all of the features which contribute to its low frequency.

Function: Operator Response During ATWS (Guideline 2.A)

Guideline 2.A. Operator Response During ATWS

Basis: The criteria developed here are based on the assumption that each of the plants is (or will be) in compliance with the ATWS final rule dated July 26, 1984. PRA studies have shown that the predicted core-damage frequency because of ATWS is significantly lowered based upon modifications which comply with the ATWS rule. The major thrust of the ATWS rule is on the addition and/or upgrading of scram related systems and equipment to prevent an ATWS. Human reliability studies performed in support of NUREG-1150 point to potential benefits for improved operator training and procedures to mitigate the effects of an ATWS and prevent core damage from occurring. For any individual plant which may be found to be vulnerable to ATWS, the following criteria reflect additional measures that emphasize the operator's role and function in mitigating an ATWS initiator.

During an ATWS sequence, the operator is required to inhibit initiation of automatic safety systems and attempt to manually control and mitigate the outcome of the event. In contrast, most other accident sequences are prevented or mitigated by systems which allow the operator to monitor automatic system initiation and require intervention only when a system fails to function adequately. Thus, an ATWS sequence requires operator responses that are in opposition to the highly trained responses required for the recovery and mitigation of all other off-normal and accident events. Therefore, operator training and procedures for the ATWS sequences should specifically prepare operators to perform the unique actions as well as in the other measures below.

Criteria:

2.A.1. Operator training and emergency procedures should specify the plant parameters that are indicative of ATWS and the actions to be taken to verify that the reactor coolant recirculating pumps have tripped automatically. Additionally, they should specify the actions to be taken if the reactor coolant recirculating pumps do not trip automatically.

Table 4.2 (Continued)

2.A.2. Operator training and emergency procedures should ensure that standby liquid control system (SLCS) injection is initiated, as required, during an ATWS. Operator training and procedures should also specify the plant parameters that indicate SLCS actuation and the actions to be taken and verification to be made to ensure that the SLCS was actuated.

Note: PRAs which have investigated plants with automatic initiation of two train SLCS, alternate rod insertion and high capacity SLC boron injection systems have found greatly reduced ATWS frequency (less than 10^{-7} per year).

2.A.3. Operator training and procedures should ensure operator familiarity with reactor pressure vessel (RPV) water level and flow control during ATWS.

Note: This unique control requires actions which conflict with mitigating actions for all other accidents which call for flooding the RPV to ensure the reactor core is covered.

2.A.4. Since RPV water level indicators may be inaccurate and may provide conflicting indications of the water level, operator training and procedures should provide guidance to the operator.

2.A.5. The automatic depressurization system (ADS) should be capable of being overridden by the operator during an ATWS before its automatic initiation. Operator training and procedures should address the possible reluctance of operators to defeat a safety system, in particular, the need to inhibit the ADS immediately after an SLCS initiation. (If the SLCS is unavailable, depressurization may be required.)

2.A.6. Operator training and procedures should specify the responsibilities of operating staff crew members and clarify how information will be exchanged among them. In particular, instrumentation readings may have to be relayed between the crew member(s) operating the control boards and the senior reactor operator coordinating the crew's response to the accident.

2.A.7. The capability of the ATWS response-required systems and equipment to function under predicted environmental and fluid loads associated with severe-accident sequences should be assessed to determine whether the equipment would be available for an appropriate time of operation.

2.A.8. An assessment should be made of the feasibility of establishing the main condenser as a heat sink by reopening the main steam isolation valves and turbine bypass valves, if possible.

Table 4.3 Criteria for BWR Mark III Containment
Guideline 3: Station Blackout

Concern:	Station blackout sequences have been shown to be one of the leading classes of severe-accident sequences both in terms of core-damage frequency and risk.
Functions:	Reactor Pressure Vessel (RPV) Injection (Guideline 3.A) Hydrogen Control (Guideline 3.B)

Guideline 3.A. Provide Reactor Pressure Vessel Injection

Basis: Significant study and research have preceded current work on severe accidents, in particular, reference is made to the rule-making activity already under way on station blackout. It is assumed that when the station blackout rule is finalized, some requirements of the rule may be similar in form to the criteria below. Nevertheless, during an individual plant examination (IPE), it is important to highlight those areas which previous PRAs found to be important contributors to the station blackout core-damage frequency. For those specific plants which are found to be vulnerable to station blackout events, the criteria below will assist in identifying potential areas for plant improvements as well as identifying operator actions which are key to mitigating a station blackout event.

Criteria:

3.A.1. For the BWR-6 design, the reactor core isolation cooling (RCIC) System is intended for the purpose of RPV injection independent of ac power. However, it has been postulated that this system cannot sustain itself in the presence of a prolonged blackout. Therefore, the RCIC should be assessed with respect to its capability to function in the presence of station blackout.

3.A.2. Given that the high pressure core spray (HPCS) diesel is not part of the station blackout, the HPCS system capability of performing its intended function while under station blackout conditions should be assessed to determine whether it would be available for an appropriate time of operation.

3.A.3. Operator training and procedures should specify the plant parameters indicative of HPCS and RCIC initiation. Additionally, the training and procedures should specify the actions required to place and/or ensure that these systems are in operation under station blackout conditions.

3.A.4. Special emphasis should be placed upon the review of the ac and dc power systems to ensure that common cause failures have been eliminated to the extent practical from the design.

Table 4.3 (Continued)

Guideline 3.B. Provide Hydrogen Control

Basis: The ASEP stucy for Grand Gulf found the dominant contributor to core melt to be station blackout. For this type of sequence NUREG-1150 predicts that sufficient hydrogen will be produced to threaten the containment integrity because of hydrogen detonation or deflagrations. The present hydrogen control system is ac dependent and will not be available during these sequences.

Criteria:

3.B.1. Operator training and procedures should specify methods and actions to prevent initiation of the hydrogen control system under conditions which may lead to a hydrogen detonations.

3.B.2. The capability of the hydrogen control system to function under predicted environmental and fluid loads associated with station blackout should be assessed to determine whether the equipment would be available for an appropriate time of operation.

Guidance: A suitable hydrogen control system would have a dedicated power supply system to preserve its function for the anticipated hydrogen generation phase of a severe accident resulting from station blackout.

Table 4.4 Criteria for BWR Mark III Containment Guideline 4: Loss of Containment Heat Removal Sequences

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Concern: Failure to remove the decay heat buildup in the suppression pool (loss of containment heat removal) following a transient event has been shown to create net positive suction head (NPSH) problems for some ECCS pumps taking suction from the suppression pool and, therefore, could lead to injection failure, subsequent core damage, and containment failure. The RSS indicated that this was a leading class of core-damage sequences for a Mark I plant.

Function: Emergency Core Cooling (Guideline 4.A)

Guideline 4.A. Provide Long-Term Emergency Core Cooling (ECC)

Basis: Implementation of the following criteria will ensure that the failure potential of ECC injection because of loss of the containment heat removal function remains low for all Mark III plants.

Criteria:

4.A.1. If suppression pool temperature precludes the use of primary (ECC) injection paths, operator training and procedures should specify methods and actions for heat removal via specified alternate injection path(s).

4.A.2. For the alternative injection path(s), it should be demonstrated that the flow would be sufficient to preclude core damage.

4.A.3. If local operation of any equipment is required, the time required to perform these functions should be consistent with the time available to help prevent core damage and account for personnel exposure to the predicted severe-accident environment.

4.A.4. The capability of equipment used for alternate injection to function under the predicted environmental and fluid loads associated with severe accident sequences should be assessed to determine whether the equipment would be available for an appropriate time of operation.

Table 4.5 Criteria for BWR Mark III Containment Guideline 5: Support System Interdependencies

Concern: When conducting a PRA, IPE or similar analysis, it is imperative that the support system interdependencies be fully developed, understood and reflected in the final results. Otherwise there is no assurance that the dominant core-damage/risk sequences have been identified.

Function: Support System Interdependencies (Guideline 5.A)

Guideline 5.A. Examine Support System Interdependencies

Basis: Implementation of the following criteria will help to ensure that the full set of support system interdependencies have been identified and have been reflected in the results.

Note: The following criteria are easily outlined but are not easily implemented. The complex nature of a nuclear power plant makes it imperative that this area of analysis be fully examined. However, since no two plants have identical support systems this analysis can only be done on a plant-specific basis.

Criteria:

5.A.1. All systems that provide any direct support to either a frontline or support system should be identified along with its supported system. For each dependency that is identified, the failure mechanism and time should be estimated.

5.A.2. Each dependency should be conditioned as appropriate as to what sequences or under what (if not all) circumstances it applies. In view of its importance, a separate station blackout dependency table should be provided which gives the available systems, their anticipated survival period, and the ultimate cause (e.g., no room cooling) of their failure.

5.A.3. The dependencies should then be linked together (preferably by computer) within the analysis in order that the extent to which their influence reaches through the systems to a consequence will be discovered. This will help to identify secondary dependencies to ensure that no one failure in a support system has an unknown critical outcome on other support or front-line systems.

Table 4.6 Criteria for BWR Mark III Containment Guideline 6: Maintain Containment Integrity

Concern: Breach of the containment boundary in the progression of a severe accident can lead to significant releases of radioactivity.

Functions: Containment Venting of Noncondensable Gases - (Guideline 6.A)

Guideline 6.A. Provide Containment Venting

Basis: Implementation of containment venting will significantly reduce the potential for loss of containment integrity because of overpressurization events.

Caution: Containment venting should not be indiscriminately performed. clear understanding of the accident sequence in progress should have been assessed prior to initiating venting. The effects of venting should have been assessed and made known to the operators during the training program. The assessment should include the effects of containment venting on the operation of ECC injection systems and health consequences.

Criteria:

6.A.1. For accident sequences where containment venting has been assessed to be beneficial, containment venting should commence, except for a station blackout, when containment pressure reaches the predetermined containment venting pressure set point. In selecting the containment venting pressure setpoint, the following functions should be ensured:

a. The ultimate containment pressure capability would not be exceeded.

b. The backpressure acting on the safety relief valve assemblies would not prevent them from performing their function.

c. The vent valve assemblies would not be prevented from performing their function.

During a station blackout, containment venting should commence in accordance with the criteria developed using the BWR Emergency Procedure Guidelines (EPG), i.e., following the onset of the transient and before depletion of the station batteries. If station batteries are not available, the capability of manual initiation of containment venting should be assessed (see Criterion 6.A.2).

6.A.2. If manual initiation of containment venting is deemed necessary, the time required to perform this function should be taken into account in the training and procedures to preclude the potential for exposing personnel to the harsh environment. Otherwise, the containment venting valve(s) should be capable of being remotely actuated during a station blackout.

Table 4.6 (Continued)

6.A.3. Operator training and emergency procedures should specify the plant parameters that will prompt the operators to make preparation, commence and terminate the venting sequence. The training and procedures should also be consistent with the required actions and timing of those actions so venting will commence immediately when required (see Criteria 6.1 and 6.2). The training and procedures should further specify the flowpath(s) available for venting, specific components to be aligned, and the required positions/states for these components. The training and procedures should specify how to proceed if it is not possible to terminate venting.

6.A.4. For each accident sequence where venting is credited (e.g., assumed to prevent containment failure) the capacity of the vent lines and associated vent valves should be assessed to determine whether the venting capacity has the capability to decrease containment pressure.

6.A.5. The criteria for filtering are dependent on the potential for bypassing the suppression pool. Whether the suppression pool is bypassed or not, the radiological release should be reduced by an order of magnitude compared to no filtering. The venting flowpath should ensure that all media to be vented pass through the suppression pool thus providing filtering by the pool. If the potential of a vent path to bypass the suppression pool is high, a filter should be provided in this vent path with the ability to reduce radiological releases by an order of magnitude.

6.A.6. Equipment designated to support containment venting should be assessed for its ability to function reliably for a sufficient period under the predicted environmental and fluid loads associated with venting commencement pressure. If necessary, it should be enhanced to include operation during the vaporization release phase of core-concrete interaction.

6.A.7. The effects of possible hydrogen burn, radiation and/or steam on equipment located in the reactor building outside of the primary containment should be considered in the venting assessment. If equipment important to the mitigation of accident sequences is jeopardized by venting, alternate venting paths, judged not to be detrimental, should be identified and assessed. Consideration should also be given to the effectiveness of the reactor building blowout panels and fire sprays to accommodate the discharge through the primary containment vents, thereby ensuring reactor building structural integrity.

6.A.8. The effects of possible containment depressurization on the NPSH of the emergency core cooling related pumps should be assessed. Alternate injection sources which are unaffected by venting should be considered.

Table 4.6 (Continued)

6.A.9.	The capability to terminate venting and the conditions under which venting would be terminated should be considered in the venting assessment. Specifically, the level of radioactivity in the wetwell airspace should be considered with regard to the projected offsite consequences.

Table 4.7 Criteria for BWR Mark III Containment Guideline 7: Flooding Within the Reactor Building

Concern: An excessive water release into a portion of the reactor building (RB) outside the primary containment which houses a concentration of safety related equipment raises the possibility of a common-mode event disabling all the equipment in the compartment. At least one plant with a Mark II containment has been identified in which the location of safety equipment, including all ECCS pumps, in the lowest RB level makes the flooding initiator a substantial contributor to CDF.

Function: Prevent or Mitigate RB Flooding (Guideline 7.A)

<u>Guideline 7.A. Prevent or Mitigate Reactor Building Flooding</u>

Basis: Implementation of the following criteria may reduce the potential of a common-mode failure of safety equipment because of RB flooding in Mark III containments where RB layout combines important safety equipment in low-lying portions of the RB with exposure to possible inundation.

Criteria:

7.A.1. Operator training and procedures should ensure that the operator will diagnose and isolate any flooding of the RB that occurs.

7.A.2. Operator training and procedures should ensure that the operator is prepared to use alternate injection sources still available if flooding causes a common-mode failure of ECCS equipment.

7.A.3. The electrical systems should be assessed for the possibility of cascading failures because of flood induced electrical shorts. Additional isolation devices (e.g., circuit breakers) should be considered, if needed.

APPENDIX A

SEVERE-ACCIDENT RISK INSIGHTS

This appendix considers various studies of BWR-6 reactors with Mark III containments (BWR Mark III). The paradigm chosen for the present purposes is the Grand Gulf design. Insights from selected studies that contributed to the development of the specific guidelines (Section 3) for the prevention and mitigation of severe accidents are identified and discussed.

The approach used here, to characterize the BWR Mark III risk profile, employs the Peach Bottom (BWR-4, Mark I) analysis[1] as a stepping stone. As Peach Bottom was the first plant analyzed in this severe accident program, an extensive analysis and comparison of the various studies was performed and documented not only to identify the important areas of risk for Peach Bottom but also to form the baseline analysis for the plant analyses that followed. After comparing the various BWR Mark I related studies, it was concluded in that analysis that most of the studies pointed to the same key features as being important. Based on this insight, the focus of this effort was to determine the dominant sequence types for BWR Mark III plants by applying a set of screening criteria given in Section 3 and to identify the associated containment failure modes.

A.1 Core-Damage Profile

The core-damage profiles from a number of BWR Mark III plant studies have been compiled in Table A.1. These studies include the Grand Gulf Reactor Safety Study Methodology Application Program (RSSMAP)[2] and the Grand Gulf Industry Degraded Core Rulemaking Program (IDCOR)[3] analyses and the results found in the BNL Review[4] of the General Electric Standard Safety Assessment Report (GESSAR) II PRA.[5] Explicit references to the GESSAR II work itself have been avoided because of its proprietary nature.

A second set of core-damage profiles relating specifically to Grand Gulf are found in Table A.2. This table was constructed in the attempt to reconcile the presentation of the RSSMAP and IDCOR results from Table A.1 with the format and content of the Accident Sequence Evaluation Program (ASEP)[6] study of Grand Gulf. This reconciliation process was performed for the following reasons: (1) ASEP explicitly presented station blackout sequences and these overwhelmingly dominate the estimated core-damage frequency (CDF) whereas the two other studies have station blackout cut sets within other sequences and, (2) the ASEP results were not grouped by initiator into the two categories of T_1 (loss of offsite power) and T_{23} (all other transients) as were the other two studies.

For the RSSMAP study,[2] all of the leading cut sets presented for the T_1 initiator in the report were examined to identify the contribution of station blackout conditions (i.e., loss of both offsite power and diesel generators 1 and 2). All cut sets displaying these conditions were subtracted from their original sequences and added together to form a station blackout sequence. The remaining (non-blackout) sequence contributions (T_{23}) were then combined with the remaining T_1 contributions and presented in Table A.2.

Leading sequence cut sets were not provided in the IDCOR analysis. However, Figure A-18[3] does provide the station blackout contribution within the T_1QUV sequence. Therefore, T_1QUV was reduced by the station blackout (TB) portion and combined with T_{23}QUV in Table A.2. As no other station blackout contributions were identified, the remaining like sequences were combined (T_1 + T_{23}) and presented in Table A.2 along with the derived TB contribution.

The following subsections will address the leading CDF sequences including appropriate comparisons between the referenced studies.

Station Blackout

From Table A.2 it can be seen that station blackout (SBO) is relatively significant in all three Grand Gulf studies. SBO is defined as loss of offsite power coupled with failure of the Train 1 and Train 2 emergency power systems. Loss of the Train 3 (HPCS) diesel generator/emergency power system is not required in order to ensure a core melt assuming no recovery of the other power systems.

The ASEP study focused directly upon SBO and modeled SBO explicitly within the event trees. Figure A.1 presents the five leading blackout sequences which totally dominate the calculated ASEP CDF. From Figure A.1 it can be seen that dependent failure of the HPCS (U_1) and reactor core isolation cooling (RCIC) (U_2) systems are key contributors to these sequences. Both short-term sequences (3 and 5) include the independent failures of both HPCS and RCIC, and the long-term sequences show RCIC or HPCS as an initial success and yet all five sequences shown in Figure A.1 result in core damage and containment failure. This is because of the modeling assumptions applied to the HPCS and RCIC systems. The HPCS and RCIC systems have been modeled with three time-dependent failure modes in the blackout sequences. These are failures because of (1) loss of dc control power because of battery depletion in 11-12 hours (this only applies to HPCS if its dedicated diesel generator has also failed), (2) pump seal failure in a 6-8 hour period based upon temperature rise in the suppression pool, and (3) failure because of loss of room cooling in about twelve hours. Because of the shorter time to failure, pump seal failure is dominant.

Table A.3 shows the possible effects of removing these time-dependent blackout sequence failure modes of HPCS and RCIC. The results of this assumption show that short-term failures would then represent the blackout CDF contribution and the overall total CDF contribution would be reduced, in this example, by almost an order of magnitude.

In the IDCOR analysis, the HPCS and RCIC systems are not modeled as guaranteed failures in the long term, given failure to recover an ac power source, as was discussed above. Rather, the IDCOR analysis assumes that the batteries will deplete in about five hours (versus the 11-12 hours in ASEP) and this failure will therefore occur before the seal failure assumed in ASEP. Battery depletion therefore may be a more benign dependent failure mode as it leaves the HPCS and RCIC systems (pump seals) undamaged and available for subsequent recovery of ac power. In summary, the assumed time to battery depletion is the overwhelming driving factor between the difference in quantification between IDCOR and ASEP. It is clear that no matter which of these two studies more accurately reflects the present Grand Gulf plant, the key to reducing the

SBO contribution to CDF is not in simply adding battery capacity, but rather in providing long term protection to the pump seals.

In addition, the results of both the GESSAR II PRA[5] and the BNL review[4] of that document are essentially in agreement with the Grand Gulf ASEP study in that loss of offsite power (LOOP) (predominantly SBO) events are the dominant contributors to core damage.

Anticipated Transients Without Scram (ATWS)

The CDF resulting from an ATWS event has been significantly reduced from the RSSMAP study to the IDCOR "committed" results and even further in the ASEP results (Table A.2). The contributing factors to this are as follows. First, within the IDCOR analysis, the addition of an alternate rod insertion (ARI) system has effectively decreased the scram failure frequency by a factor of three. Another factor is the doubling (from 43 to 86 gpm equivalent) of the boron flow of the standby liquid control (SLCS) system. This has the effect of allowing more time to elapse before SLCS must be activated. SLCS failure in the RSSMAP study was dominated by operator failure to actuate. Therefore lower operator failure probabilities are used in the IDCOR analysis to reflect the additional time available. Additionally, there now exist emergency procedure guidelines that facilitate the operators dealing with an ATWS event including the possibility of depressurizing and using the low pressure injection systems.

The ASEP ATWS analysis does not include any additional hardware upgrades beyond that of the IDCOR analysis. It does however, appear that a more detailed analysis has been performed of the probability of operator error including an assessment of the likelihood of success for low-pressure injection. These two areas apparently account for the further reduction in ATWS CDF calculated. Specifically, a more rigorous look at operator actions given detailed ATWS procedures and training has yielded lower human error probabilities (HEPs) in the ASEP analysis. Also, the resulting lower HEP associated with actuating both trains of SLCS within ten minutes is assumed to keep the suppression pool temperature at or below 180°F. This results in a lower probability of failure of the HPCS pump seals, thus reducing the probability of one of the failure modes for high-pressure injection.

The ATWS rule assumes that BWR-6 designs which incorporate the hardware features noted above need not make additional significant hardware modification to keep ATWS core-damage probabilities low. As can be seen from the results of the ASEP and IDCOR analyses, further reduction in CDF is still possible by upgrading operator performance. Based upon this result, operator actions in response to an ATWS event are addressed in one of the guidelines to assure that other BWR-6's do not have a high ATWS CDF. It is further noted that the GESSAR II PRA Review had comparable results to the IDCOR analysis (See Table A.1).

Loss of Containment Heat Removal

In the Grand Gulf RSSMAP study,[2] the phenomenon of loss of containment heat removal dominated the CDF. In the later studies, these types of sequences no longer make a significant contribution. The RSSMAP analysis included the assumption that containment failure resulted in loss of injection

capability. According to the ASEP study: "It has been determined that deformation of injection lines does not occur, and since the systems that take suction from the suppression pool can pump saturated water, loss of injection does not occur as a result of containment failure."

In addition to the change in the assumption of injection failure noted above, the IDCOR and ASEP studies also investigated ways of delaying or preventing containment failure itself which were not accounted for in the RSSMAP study. These mechanisms include venting the containment and the use of alternate injection paths. When these alternate success criteria are factored into the Grand Gulf model, as was done in the IDCOR and ASEP analyses, the estimated CDF for loss of containment heat removal sequences is reduced to less than 10^{-7} per reactor year. In accordance with these insights, a guideline has been provided in Section 4 to ensure that other BWR Mark III designs incorporate emergency procedures that will ensure that loss of containment heat removal sequences are kept at a similarly low level.

Interfacing Systems LOCA

Traditionally, interfacing system LOCAs have been identified in PRAs as low frequency but high consequence events. The Grand Gulf studies do not identify these sequences as high risk events. It appears that the basis for this result stems from the very low to negligible probabilities calculated for these events. BNL is currently conducting a study of BWR and PWR interfacing system LOCAs (GI-105). It is recommended that this issue not be dropped from the severe accident guideline list (as would be suggested by the results of the Grand Gulf studies) pending resolution of this generic issue.

A.2 Core-Meltdown Phenomena and Containment Response

In the previous section important core-meltdown accident sequences were identified in terms of the overall core-melt frequency. In this section, a review of the core-meltdown phenomena and containment response appropriate to these accident sequences is presented. In addition, accident sequences are examined which, although they do not appear to be important to the overall core melt frequency, may pose a unique or very severe threat to containment integrity. The review will again rely heavily on the IDCOR and SARP[6] analyses which were specifically carried out for the Grand Gulf plant. The review also will take into account other studies pertinent to a BWR-6 with a Mark III containment and, in particular, the Containment Loads Working Group (CLWG) report[7] and the Containment Performance Working Group (CPWG) report.[8]

A typical Mark III containment building is shown in Figure A.2. The Mark III containment relies on water to condense any steam that might be released from the primary system during an accident. Containments of this design are called pressure-suppression containments. Mark III containments are very effective at condensing steam but they may be vulnerable to buildup of combustible and noncondensable gases that would be generated during a severe core-meltdown accident.

The aim of this section is to identify severe-accident threats to the containment appropriate to the accident sequences identified in Section A.1. These threats are then used to determine the most probable modes of containment failure. These, in turn, identify the potential release paths for

fission products to reach the environment. This section, therefore, provides the link between the identification of core-meltdown accident sequences and the determination of fission-product release paths.

Reference 9 is a Battelle Columbus Laboratories (BCL) study of the Mark III containment responses to the two leading core-damage sequences, i.e., blackout and ATWS. The blackout sequences have been divided into two scenarios and the ATWS sequences are grouped into one scenario.

One blackout scenario assumes that containment failure follows soon after vessel failure because of a hydrogen explosion. The other blackout scenario assumes that a hydrogen explosion does not occur either because of a slow burning rate or the lack of an ignition source. The long term failure of the containment then occurs as the result of overpressurization because of noncondensable gas generation. In both scenarios it is assumed that there is leakage that bypasses the pool. The amount of this leakage has a direct bearing on the magnitude of the radioactivity released and is assumed to be much greater for the short-term hydrogen-explosion scenario.

The ATWS sequences are characterized by containment overpressure failure prior to core damage. The containment is overpressurized because of power generation greater than that which can be removed by the residual heat removal system. This causes the suppression pool to heat up and pressurize the containment to failure in a rather rapid fashion. The high pool temperatures are assumed to cause pump seal failure resulting in loss of the emergency core cooling systems and thus core damage.

The Grand Gulf studies also point to a possible suppression pool bypass mechanism in which corium ejected from the vessel may erode the pedestal wall causing displacement of the vessel which in turn may disrupt penetrations through the drywell wall with the possibility of pool bypass. The IDCOR study investigated the addition of a drywell spray system to prevent this situation. The IDCOR analysis stated that a drywell spray system can be used to reduce drywell pressure, cool the drywell, and quench the melt and reduce radioactivity in the drywell in case of a core melt accident. It further stated that the system could preserve containment integrity during an accident by quenching the corium and reducing the likelihood, size and consequences of a potential release. The IDCOR analysis concluded, however, that this modification was not necessary since the probability of pool-bypass sequences was estimated to be low even without the drywell spray system.

The results of an assessment of core-meltdown phenomena and containment response is usually expressed in terms of a containment matrix. A containment matrix provides the framework for estimating the conditional probabilities of a particular accident sequence resulting in a variety of containment failure modes (or fission-product release paths). The IDCOR and Severe Accident Risk Reduction Program (SARRP) containment matrices are given in Table A.4. From an inspection of Table A.4 it is clear that the SARRP approach includes a higher potential for drywell leakage and pool bypass than the IDCOR approach. Differences in the probabilities in Table A.4 are because of differences in modeling assumptions for core meltdown and containment response in the IDCOR and SARRP studies. These differences are discussed in detail in the following sections.

The review of the IDCOR and SARRP analyses of core-meltdown phenomena and containment response was greatly assisted by the IDCOR/NRC meetings that have been held specifically to identify differences between the approaches adopted by the two groups and to develop a way of resolving these differences. These meetings identified 18 broad NRC/IDCOR issues that highlight significant differences between the approaches of the two groups. These issues are listed in Table A.5 but they do not all apply to a BWR Mark III and some are not related to core meltdown phenomena and containment response. Of the 18 issues, 8 have been identified that are pertinent to the subject of this section. Each issue, is discussed, in turn in the following sections. Differences between IDCOR and SARP will be identified and their significance will be indicated.

A.2.1 In-Vessel Hydrogen Generation (NRC/IDCOR Issue 5)

There are significant differences between the IDCOR and SARP predictions of hydrogen (H_2) generation during in-vessel core melting. During the early stages of core heatup and degradation (while the fuel rods are still in place in the core region), both IDCOR and BCL predict similar H_2 generation. However, after the fuel rods and cladding begin to melt and relocate into the bottom of the reactor vessel, the BCL analysis with the Source Term Code Package (STCP) indicates more H_2 generation than the IDCOR analysis.

Hydrogen is important to containment loading because it is a combustible and noncondensable gas. The Mark III containments are not inerted but are required to have igniters installed to control burning of any H_2 that may be released. However, H_2 combustion (Issue 17) could be a threat to the Mark III containments if the igniters fail to burn the hydrogen gradually. For instance, recovery of ac power after an initial blackout could lead to activation of the igniters with dangerous levels of hydrogen existing in the containment, according to the SARRP analysis. Mark III containments are also susceptible to the long term buildup of noncondensable gases (such as H_2 and carbon dioxide) which could threaten containment integrity by overpressure. The larger amount of H_2 generated in-vessel in the BCL analysis leads to a higher predicted containment pressure prior to vessel failure than in the IDCOR analysis. BNL staff have performed an extensive assessment[10] of in-vessel H_2 generation including particularly with regard to accidents that resulted in core damage but which were terminated by subsequent coolant injection. The results of these calculations indicate the potential for more H_2 generation than predicted by IDCOR.

The differences in H_2 generation were found[1] to have very little impact on risk for the Mark I containment since it is inerted. However, the Mark III containment is not inerted and the difference in hydrogen generation appears to have an important effect on risk. This is particularly so since the only hydrogen control device (igniters) will not function during the dominant coremelt sequences (station blackout).

A.2.2 Core Slump, Core Collapse, and Reactor Vessel Failure (NRC/IDCOR Issue 6)

This is another area in which there are significant differences between the IDCOR and the ASEP/SARRP analyses. The importance of these differences to overall risk again depends on plant specific systems. Section A.2.1 indicated that the predicted hydrogen generation during core slump was quite different

in the IDCOR and BCL analyses. The larger hydrogen generation contributes to a larger probability of significant drywell to wetwell leakage (.42) for SARRP (refer to Table A.4).

The core slump and reactor vessel failure models also significantly influence the initial conditions for ex-vessel interactions of the core debris with water or concrete. The IDCOR core slump model assumes that after 20% of the core has melted, it will relocate into the bottom of the reactor vessel, which will then rapidly fail because of local penetration failure. Thus only a relatively small fraction of the core will be initially released from the reactor vessel. The remainder of the core melts down over a much longer time period. A similar philosophy has been adopted in the draft NRC staff issue paper on direct heating. This work states that the BWR core support design (which provides individual support for each group of four fuel bundles from the vessel bottom head) is judged to minimize the probability of high pressure ejection of a large fraction of core debris into the containment. Slumping of relative small quantities of core debris (because of localized failure of the supports) is anticipated to result in depressurization of the vessel (because of local melt-through) before large quantities of molten core material have collected in the bottom head.

On the other hand, the STCP core slump model used in the BCL analysis assumes total collapse of the core into the bottom of the reactor vessel after 75% of the core is predicted to melt. Thus, a large fraction of the core debris is available to be released when the vessel is predicted to fail in the BCL analysis. The much larger quantity of core materials released from the vessel at the time of vessel failure in the BCL analysis has important implications for the Mark III containment. If the primary system is at high pressure during core meltdown, then the molten core materials will be ejected under pressure from the reactor vessel when it fails. Section A.2.4 discusses the phenomena that could occur when molten core debris is ejected from the reactor vessel under pressure. Since more core debris is predicted to be ejected, the resulting pressure/temperature loads in containment will be correspondingly higher.

Sandia National Laboratories (SNL) has also performed an uncertainty study in support of NUREG-1150 which examines the range of possible core slump behavior and attaches a low likelihood to the high core-melt fraction slump model.

If the primary system is depressurized during core meltdown, then the core debris will fall into the region below the reactor vessel after it fails. Obviously, if more core debris is predicted to fall into the pedestal region, then the resulting molten pool will be deeper and there will be a greater potential for the core to erode the support pedestal and possibly fail the drywell wall. SNL has identified this as a mechanism for pool bypass after vessel failure with a conditional probability of approximately .02.

From the above discussion, it is clear that differences between the IDCOR and BCL/SNL analyses for core slump and vessel failure are significant. The potential for early containment failure depicted in the IDCOR and SNL containment event trees (refer to Table A.4) is in substantial agreement in spite of these differences. However, the effect of phenomenological uncertainties has not been addressed yet. The authors concur with IDCOR and SNL in attaching a

small probability to early failure of the drywell wall because of contact with core debris which would result in large releases.

A.2.3 Containment Failure Because of In-Vessel Steam Explosions (Issue 7)

The potential for an in-vessel steam explosion to occur and generate a missile capable of failing containment was investigated by a group of experts and the results were published in NUREG-1116.[11] The conclusion of this expert group was that the occurrence of such an event has a relatively low probability. These results are reflected in the SNL and IDCOR containment event trees. The allocation of a very low conditional probability (10^{-4} per reactor year) of occurrence to this event is supported by the authors.

A.2.4 Direct Heating of Containment (Issue 8)

This is an area of significant phenomenological uncertainty related specifically to core meltdown with the primary system at high pressure. If molten core materials are ejected from the reactor vessel under pressure, experiments[12] at SNL have indicated that they form fine aerosols, which are dispersed into the containment atmosphere and directly heat it. An additional concern is the oxidation of the metallic content of the core debris. These reactions are very exothermic and would add an additional heat load to the containment.

The pressure rise in containment because of direct heating is directly proportional to the quantity of core debris dispersed from the reactor vessel. Section A.2.2 noted that the BCL analysis predicts significantly more debris release at vessel failure than the IDCOR analysis. Thus the potential for early containment failure because of direct heating is higher in the BCL analysis. However, the SNL event trees attach a high probability to a slow melt release from the vessel and thus failure because of direct heating is low.

The assumption that all the core debris is released at vessel failure (BCL analysis) is clearly conservative. The IDCOR and SNL analyses appear to be too optimistic considering the lack of supporting large scale experiments. In addition to the pressure loads imposed by the dispersed core materials, there is the concern that the hot core debris could erode the support pedestal and fail it (see Section A.2.2).

A.2.5 Ex-Vessel Heat Transfer Model From Molten Core to Concrete (Issue 10)

This issue is of concern to Mark III containments because heat transfer from the top of molten core materials (on the drywell floor) directly heats the drywell atmosphere. Thus, differences in heat transfer from the top of the core debris can result in significant differences in the predicted drywell atmospheric pressures and temperatures. The IDCOR model[13] transfers more heat from the top of the core debris than the STCP model (CORCON Mod 2).[14] Thus, IDCOR predicts higher drywell temperatures than the BCL analyses. However, because IDCOR predicts high heat transfer from the top of the core debris, the concrete erosion velocities are much lower than the BCL predictions. Lower concrete erosion results in less gases and aerosols released from core-concrete attack and thus lower pressures in containment.

Differences in the predicted drywell pressure and temperature histories can influence the potential for suppression pool bypass (Issue 13A) and containment performance (Issue 15).

A.2.6 Suppression Pool Bypass (Issue 13A)

If the fission products pass through the suppression pool both IDCOR and BCL predict significant retention of fission-product aerosols in the water. The amount of retention depends on several factors such as submergence, water temperature, aerosol particle size, carrier gas composition, and others. The ability of the Mark III suppression pool to trap aerosol fission products is an important mitigative feature. Thus, any pathways that might open, which would allow the fission products to bypass the pool are very undesirable. The following are possible ways in which the suppression pool may be bypassed:

- failure of the drywell wall because of hydrogen explosions
- failure of vacuum breakers between the drywell and wetwell
- failure of drywell penetrations because of high temperature
- failure of the pedestal wall as a result of contact with molten core materials

Because of the importance of the suppression pool as a mitigative feature, the vulnerability of a Mark III containment to any of the above bypass pathways must be carefully assessed. The probability of degradation of the drywell penetrations because of high temperatures in the SNL analysis reflects much of the work of the CPWG, which had significant BNL input. Failure of the pedestal wall as a result of contact with molten core materials and the resulting displacement of the vessel is an area of great phenomenological uncertainty. Preliminary event trees from SNL indicate that substantial leakage of approximately 1 square foot through the drywell wall will occur after failure of the pedestal. The authors are unable to rule out pedestal failure as a potential cause of pool bypass, but the capability of the upper pool to dump into the drywell and quench the core appears to be an important mitigative capability in Grand Gulf.

A.2.7 Containment Performance (Issue 15)

The response of a Mark III containment to severe-accident loads is uncertain. In Section A.2.5, it was noted that IDCOR predicts very high drywell temperatures but does not predict drywell failure. IDCOR assumes that a relatively small opening will occur in the outer containment which allows gradual leakage with no pool bypass. By comparison the BCL analysis with the STCP allows for primary containment failure because of overpressure and assumes an opening large enough to rapidly depressurize the primary containment. In addition, the BCL analysis allows for degradation of drywell seals because of high temperatures. Seal degradation was assumed to result in a gradual leakage from the drywell in the BCL analysis.

Differences in containment performance can influence the timing and quantities of fission products released to the environment (refer to Section A.3). However, these differences do not lead to major differences in the predicted overall risk for the Mark III containment as discussed in Section A.4.

A.2.8 Hydrogen Ignition and Burning (Issue 17)

Although there are considerable differences in the rates of hydrogen generation both IDC R and BCL predict that hydrogen burning gives a high probability of early containment failure for station blackout. Although Grand Gulf has installed igniters to prevent hydrogen detonation in compliance with the interim hydrogen rule, the igniters require ac power and are not available for the dominant class of core-melt accidents (station blackout) and may actually exacerbate the situation if power is restored later in the accident when detonable levels of hydrogen have accumulated.

A.3 Fission-Product Release

Section A.2 identified potential containment failure modes or fission-product release paths appropriate to the important core-meltdown accident sequences identified in Section A.1. The aim of this section is to determine the timing and amount of fission products released from the damaged fuel and predict the subsequent mitigation of these fission products along the release paths identified in Section A.2. The IDCOR and BCL analyses for the Grand Gulf plant are used as the basis for these calculations.

In order to review the differences in approach, IDCOR and NRC contractor analyses (performed for SARP) are compared in Tables A.6 and A.7 for ATWS and SBO sequences respectively. The IDCOR methods predict lower releases of all fission-product groups than predicted by BCL. The reasons for the different predictions in Tables A.6 and A.7 are complex but were discussed during the numerous IDCOR/NRC meetings and they are included in the list of 18 NRC/IDCOR issues in Table A.5. Out of the 18 issues, 6 are pertinent to fission product release and transport. However, not all of the 6 are major contributors to the differences in Tables A.6 and A.7. The most prominent differences are displayed in Table A.7 for SBO. The BCL analysis assumes a large leakage through the drywell wall thus bypassing the pool while IDCOR assumes that there is no pool bypass. Each of the 6 issues is discussed in the following subsections.

A.3.1 Fission-Product Release Before Vessel Failure (Issue 1)

This is one issue that does not contribute significantly to the differences between the IDCOR and BCL analyses in Tables A.6 and A.7. Both studies predict similar releases of the more volatile fission products during in-vessel core degradation with the exception of tellurium (Te). However, a recent report by IDCOR assessed the impact of the Te treatment and modeled similar in-vessel Te releases to the BCL analyses. Differences in the predicted environmental releases of Te in Tables A.6 and A.7 are therefore not because of differences in the in-vessel Te release and retention models but because of differences in the amount of fission products which are assumed to bypass the pool (also refer to Section A.3.6).

A.3.2 Fission-Product and Aerosol Retention in the Primary System (Issue 4)

Differences in the initial primary system retention predicted by IDCOR and BCL are again not too significant and differ by less than a factor of two. The important difference between the IDCOR and STCP models is that in the STCP analysis fission products retained in the primary system at the point

of vessel failure are permanently retained whereas in the IDCOR analysis revaporization of these fission products after vessel failure is modeled. This is discussed in more detail in Section A.3.4.

A.3.3 Ex-Vessel Fission-Product Release (Issue 9)

There are significant differences between the IDCOR and BCL analyses for fission product release as a result of core-concrete interactions. The higher releases of the strontium, lanthanum, and cesium groups in Tables A.6 and A.7 in the SARP analyses are because of the modeling of ex-vessel fission-product release. The potential for fission-product release and inert aerosol generation during core-concrete interactions was not modeled in the IDCOR analysis of Grand Gulf. IDCOR argued that by modeling the aerosol generation during core-concrete interactions the increased aerosol density in containment would increase aerosol agglomeration and settling, thus reducing the predicted environmental release fractions relative to those predicted without this additional aerosol source.

We do not consider that this IDCOR argument has been adequately supported. In addition, the IDCOR predicted core debris temperatures during core-concrete interactions are very high; based on experimental evidence, one would expect the release of some of the refractory fission-product groups at these temperatures. We therefore believe that IDCOR should calculate the release of the refractory fission products and the associated inert aerosols. The BCL analysis currently models the release of the refractory fission products and the inert aerosols and the environmental release fractions are not low (refer to Tables A.6 and A.7).

A.3.4 Revaporization of Fission Products from the Primary System (Issue 11)

Section A.3.2 indicated that revaporization is an area of major difference between the IDCOR and SARP analyses. SARP does not model revaporization of fission products from the primary system after reactor vessel failure whereas IDCOR does model this effect. The IDCOR revaporization model means that generally more of the volatile fission products are predicted to be released from the primary system later in the accident sequence than in the NRC contractor approach in which revaporization is not modeled. However, the IDCOR model predicts lower primary piping temperatures because of larger heat losses in a BWR than in a PWR. These lower temperatures result in substantial fission-product retention in the IDCOR model even with revaporization. In spite of the revaporization in the IDCOR analysis, the IDCOR release fraction remains small since they assume no bypass of the pool (Issue 13A).

A.3.5 Fission-Product Deposition Model in Containment (Issue 12)

This is another issue that does not contribute significantly to the differences between the IDCOR and SARP analyses in Tables A.6 and A.7. Issue 13A (suppression pool bypass is discussed in Section A.2.6) really drives the differences in these two analyses. However, this issue may be of more importance to other containment designs.

A.3.6 Secondary Containment Performance (Issue 16)

For the Mark III design the reactor building does not surround the primary containment and neither SARRP nor IDCOR has taken credit for decontamination in the reactor building. Therefore, it is not an area of major difference between the IDCOR and SARP analyses.

A.4 Offsite Consequences

In this section, the potential offsite consequences of the severe accidents described in the previous sections are examined. There is one NRC/IDCOR issue related to offsite consequences, which concerns differences in the assumed evacuation models. Differences in the evacuation model influence the predicted early health effects. The issue is largely resolved and is related to the fraction of the population assumed not to participate in the evacuation.

Table A.8 gives the consequence calculations for IDCOR and SARRP[16] for several accident sequences and failure modes. This table indicates that if the containment is predicted to fail (either early or late) and the suppression pool is not bypassed, then the offsite person-rem predictions are similar ($\sim 10^5$) for the accidents considered. The only time that a significant increase ($\sim 10^7$) in person-rem is calculated is with complete pool bypass. These results clearly show that preventing pool bypass is the key to mitigating the fission-product releases for a Mark III containment. For the BWR Mark III results IDCOR and SARRP show a low likelihood of pool bypass and substantial scrubbing of the fission products. However, the bypass scenarios tend to dominate the risk calculated in NUREG-1150 for Grand Gulf. Thus, pool bypass conditions must be avoided to ensure mitigation of fission products for the dominant accident sequences.

A.5 Summary and Risk Insights

A.5.1 Core-Damage Profile

Transients dominate the core-damage risk profile for the studies examined in Section A.1. For all of the BWR plant PRAs considered, a few sequences figure prominently in all of the respective core-damage frequency profiles. This suggests that if the probability of this relatively small subset of accident sequences can be minimized, then there is a reasonable expectation that the overall core damage frequency will be minimized. This principle is used in Sections 3 and 4 to develop guidelines and criteria to control the overall core-damage frequency (Goal 3).

It is, however, important to recognize that the qualitative accident sequence descriptors are rather general and that different hardware and/or operational failures in the various BWR Mark III plants could lead to the same general accident sequence. In order to identify the plant specific (and often times unique) potential vulnerabilities that contribute to a given general sequence descriptor (e.g., station blackout) in a given plant, a plant specific examination (such as a failure mode and effects analysis coupled with a fault tree/event tree analysis) is needed.

A.5.2 Consequence Analysis

The IDCOR assessment of core-meltdown phenomena and containment response indicates that the Mark III containment can accommodate severe accident containment loads without pool bypass. The SARRP analysis predicts substantial leakage paths from the drywell are unlikely so that any fission products which are released have the benefit of suppression pool scrubbing. The only time that a major increase in offsite consequences is predicted is if there is pool bypass. This demonstrates the importance of preventing pool bypass to mitigating of fission products (Goal 1).

A.6 References

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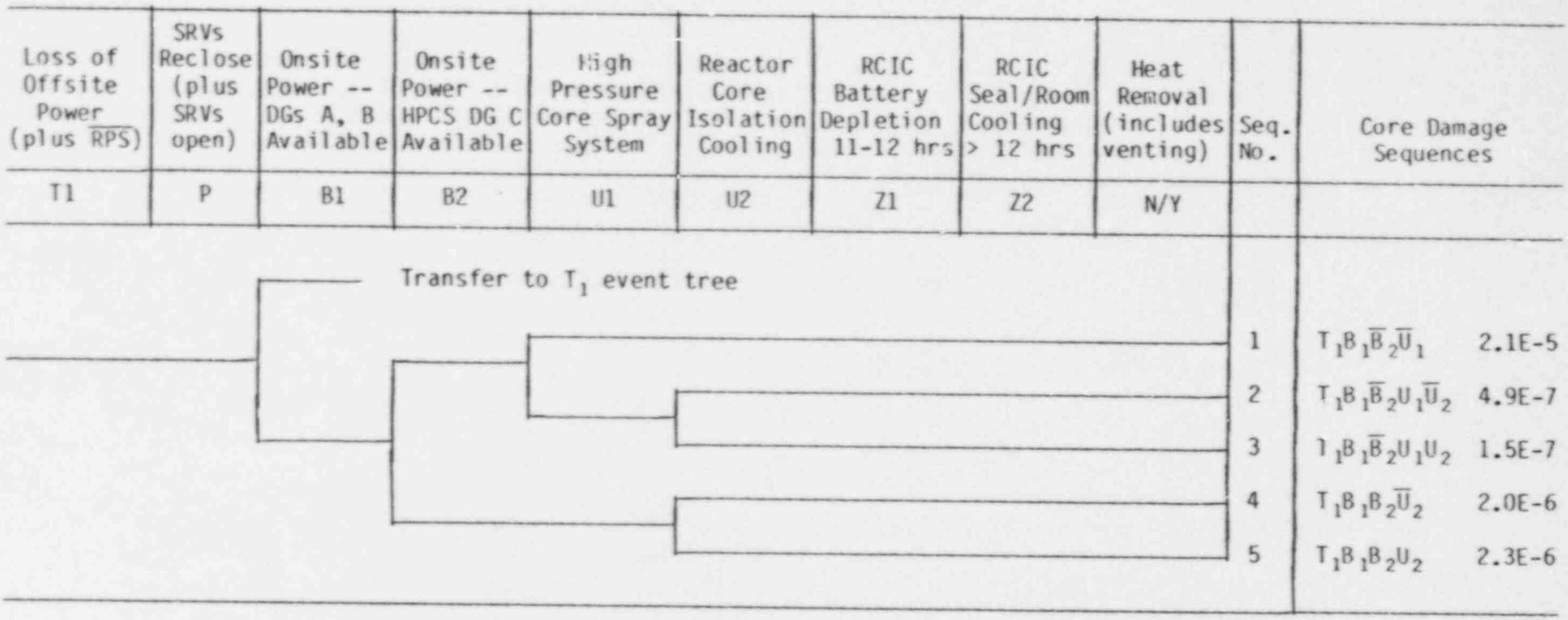

Figure A.1 Grand Gulf station blackout event tree.

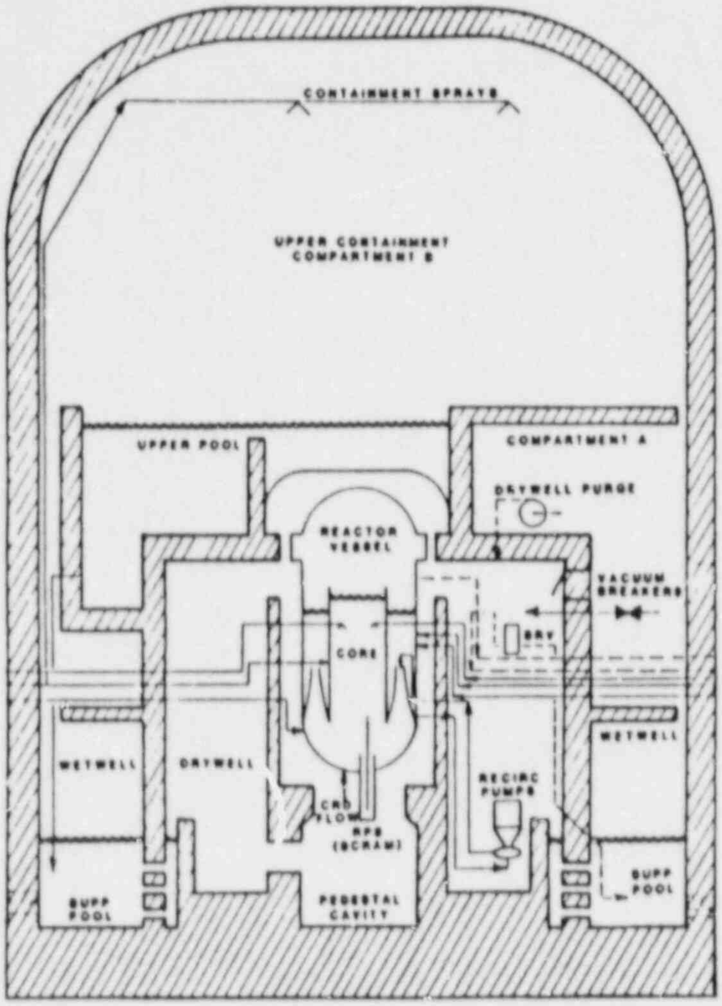

Figure A.2 Mark III containment building.

Table A.1 Selected BWR-6 Core-Damage Profiles

Sequence Type	RSSMAP	Grand Gulf IDCOR Committed	GESSAR II PRA Review
$T_{23}QW$	1.2×10^{-5}	1.9×10^{-7}	1.9×10^{-6}
T_1QW	6.2×10^{-6}	8.9×10^{-9}	--
$T_{23}C$	5.4×10^{-6}	1.2×10^{-6}	2.52×10^{-6}
SI	4.6×10^{-6}	5.5×10^{-9}	--
$T_{23}PQI$	3.7×10^{-7}	2.3×10^{-8}	--
T_1PQI	1.6×10^{-6}	2.3×10^{-9}	--
$T_1QUV + T_1QUX$	1.5×10^{-6}	4.7×10^{-7}	--
$T_{23}PQE$	5.4×10^{-7}	1.7×10^{-9}	--
T_1PQE	2.3×10^{-7}	--	--
AI	2.6×10^{-7}	1.4×10^{-9}	--
AE	5.0×10^{-9}	1.1×10^{-9}	--
$T_{23}QUV + T_{23}QUX$	5.6×10^{-8}	1.3×10^{-8}	5.3×10^{-7}
T_1C	1.2×10^{-7}	--	--
T_1QUW	3.4×10^{-8}	--	--
$T_{23}QUW$	7.0×10^{-8}	--	--
T_1W	--	--	1.6×10^{-6}
$T_1UV + T_1UX$	--	--	3.0E-5
$T_{23}UV + T_{23}UX$	--	--	5.3×10^{-7}

Table A.2 Grand Gulf Core-Damage Profile

Sequence Type	RSSMAP*	IDCOR* Committed	ASEP
TB	1.32×10^{-6}	3.4×10^{-7}	2.8×10^{-5}
TC	5.52×10^{-6}	1.2×10^{-6}	1.8×10^{-7}
TQW	1.71×10^{-5}	2.0×10^{-7}	--
TPQI	5.18×10^{-6}	2.53×10^{-8}	--
TQUV	1.42×10^{-6}	1.43×10^{-7}	$<10^{-8}$
TPQE	7.61×10^{-7}	1.7×10^{-9}	--
TQUW	1.04×10^{-7}	--	--

*These sequence core damage frequencies were derived by extracting explicit station blackout (TB) cut sets and then combining all initiators (i.e., $T_1 + T_{23}$). TB reflects the summation of the extracted cut sets.

Table A.3 Grand Gulf Station Blackout Sequence Core-Damage Frequency Point Estimates With Proposed Enhanced HPCS/RCIC Capabilities

		ASEP	Remove or Extend Blackout-Related Time-Dependent Failures[1]
Seq 1	$T_1B_1\overline{B}_2\overline{U}_1$(long term)	2.1×10^{-5}	--
Seq 2	$T_1B_1\overline{B}_2U_1\overline{U}_2$(long term)	4.9×10^{-7}	--
Seq 3	$T_1B_1\overline{B}_2U_1U_2$(short-term)	1.5×10^{-7}	1.5×10^{-7}
Seq 4	$T_1B_1B_2\overline{U}_2$(long-term)	2.0×10^{-6}	--
Seq 5	$T_1B_1B_2U_2$(short term)	2.3×10^{-6}	2.3×10^{-6}
Total	T_1B pt.est.	2.8×10^{-5}	2.45×10^{-6}
	T_1B (long term)	2.35×10^{-5} (91%)	ε (0%)
	T_1B (short term)	2.45×10^{-6} (9%)	2.45×10^{-6} (100%)

Notes:

1. In this example, removal or extension of the blackout-related time-dependent failure modes of RCIC and HPCS renders sequences 1, 2 and 4 successes and removes them from the calculated core damage frequency.

Table A.4 Comparison of the IDCOR and SNL Containment Matrices

Containment Failure Mode	Station Blackout		ATWS	
	IDCOR (T_1QUV)	SNL (TB)	IDCOR ($T_{23}C$)	SNL (TCSX)
No Containment Failure	.2	.08	--	.6
Pre-existing Leak or Failure to Isolate	.005	--	--	--
Overpressure Failure due Primarily to Hydrogen Burning	.6	.59	--	.2
Other Overpressure Failure	.2	.34	1.0	.2

<u>Drywell to Wetwell Leakage</u> (Station Blackout)	<u>SNL</u>	<u>IDCOR</u>
Design Leakage	.58	--
Late Penetration Failure	.34	--
Late Drywell Structural Failure	.02	--
Early Penetration Failure	.05	--
Early Drywell Structural Failure	.01	--

Table A.5 NRC/IDCOR Issues

Issue	Subject
1	Fission-Product Release Before Vessel Failure
2	Recirculation of Coolant in Reactor Vessel
3	Release Model of Control Rod Materials
4	Fission-Product and Aerosol Retention in the Primary System
5	In-Vessel H_2 Generation
6	Core Slump, Core Collapse, and Reactor Vessel Failure
7	Containment Failure Because of In-Vessel Steam Explosions
8	Direct Heating of Containment
9	Ex-Vessel Fission-Product Release
10	Ex-Vessel Heat Transfer Model from Molten Core to Concrete
11	Revaporization of Fission Products from the Primary System
12	Fission Product Deposition Model in Containment
13A	Suppression Pool Bypass (Pool Scrubbing)
13B	Retention of Fission Products in Ice Beds
14	Modeling of Emergency Response
15	Containment Performance
16	Secondary Containment Performance
17	Hydrogen Ignition and Burning
18	Essential Equipment Performance

Table A.6 Comparison of IDCOR and BCL Predictions of Fission-Product Release for an ATWS Sequence With No Operator Actions Taken

Event	IDCOR*	NRC Contractors**
Containment Failure (hr)	1.0	1.3
Start of Core Melt (hr)	3.0	2.0
Vessel Failure (hr)	3.8	4.2
Fission-Product Release Fractions***:		
Xe-Kr	1.0	1.0
I-Br	0.0008	0.003
Cs-Rb	0.0008	0.004
Te-Sb	0.0008	0.002
Sr	0.00001	0.002
Ba	0.00001	0.001
Ru-Mo	0.00001	Neg.
La	--	0.0001
Ce	--	0.0001

*IDCOR Technical Report 23.1GG, March 1985.
**NUREG/CR-4624, July 1986.
***Fraction of Initial Core Inventory.

Table A.7 Comparison of IDCOR and BCL Predictions of Fission-Product Release for an SBO Sequence With Hydrogen Burn

Event	IDCOR* (No Bypass)	NRC Contractors** (With Bypass)
Loss of Injection (hr)	0.0	6.0
Start of Core Melt (hr)	2.0	9.7
Vessel Failure (hr)	2.3	11.7
Containment Failure (hr)	47.0	11.7
Fission-Product Release Fractions***:		
Xe-Kr	1.0	1.0
I-Br	$7x10^{-5}$	0.016
Cs-Rb	$7x10^{-5}$	0.013
Te-Sb	$3x10^{-5}$	0.11
Sr	$1x10^{-5}$	0.3
Ba	$1x10^{-5}$	0.18
Ru-Mo	$1x10^{-5}$	Neg.
La	--	0.021
Ce	--	0.034

*IDCOR Technical Report 23.1GG, March 1985.
**NUREG/CR-4624, July 1986.
***Fraction of Initial Core Inventory.

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Table A.8 Comparison of IDCOR and SARRP Consequence Results (Person-Rem)

Accident Sequence	Containment Failure Mode	IDCOR	SARRP[16]
ATWS	Containment failure after core melt without significant pool bypass (Bin 125 Dry)	1.2×10^5	9×10^4
Station Blackout	Containment failure at RPV failure with complete pool bypass (Bin 128)	--	2 to 8×10^6
Station Blackout	Early containment failure with RPV depressurization and drywell flooding (TBS limited bypass Bin 33)	--	2×10^5
Station Blackout	Containment failure after a few hours without significant pool bypass (flooded Bin 9)	1.2×10^5	5×10^4
Station Blackout	Containment failure after a few hours without significant pool bypass (Dry Bin 3)	--	7×10^4
Station Blackout	Containment failure after many hours without significant pool bypass	2.4×10^4	--

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ASSESSMENT OF SEVERE ACCIDENT PREVENTION AND MITIGATION FEATURES:

PWR, LARGE-VOLUME CONTAINMENT DESIGN

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ASSESSMENT OF SEVERE ACCIDENT PREVENTION AND MITIGATION FEATURES:

PWR, LARGE-VOLUME CONTAINMENT DESIGN

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ABSTRACT

Guidelines and criteria have been developed for preventing and mitigating severe accidents in PWRs that have large-volume containments. The guidelines were developed from insights derived from reviews of risk assessments performed specifically for the Zion plant and from assessments of other relevant studies. Accident sequences that dominate the core-damage frequency and those accident sequences that are of potentially high consequence were identified. Vulnerabilities of the large-volume containment to severe accident containment loads were also identified. In addition, those features of a PWR with a large-volume containment, which are important for preventing core damage and are available for mitigating fission-product release to the environment were also identified. These guidelines and criteria are issued to provide direction to the analyst examining an individual plant. This direction calls attention to plant features and operator actions and provides the standards for assessing those features and actions found to be helpful in reducing the overall risk for Zion and other PWRs with large-volume containments. Thus, the guidance is offered as a resource in examining the subject plant to determine if the same, or similar, guidelines will be of value in reducing overall plant risk. These guidelines and criteria are intended to serve solely as guidance.

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NOMENCLATURE

ac	alternating current
A	medium or large loss of coolant accident (LOCA)
AFW	auxiliary feedwater
AFWS	auxiliary feedwater system
AMSAC	ATWS mitigating systems actuation circuitry
ASEP	Accident Sequence Evaluation Program
ATWS	anticipated transients without scram
BCL	Battelle Columbus Laboratories
BNL	Brookhaven National Laboratory
BWR	boiling water reactor
C	successful containment spray injection or recirculation
CCW	component cooling water
CCWS	component cooling water system
CDF	core-damage frequency
CHR	containment heat removal
CSIS	containment spray injection system
dc	direct current
E	early core melt with injection failure
ECC	emergency core cooling
ECCS	emergency core cooling systems
EPG	Emergency Procedure Guidelines
ERG	Emergency Response Guidelines
ESW	emergency service water
F	successful containment fan coolers
F&B	feed & bleed
GI	generic issue
HPI	high-pressure injection
HPIS	high-pressure injection systems
HPRS	high-pressure recirculation system
IDCOR	Industry Degraded Core Rulemaking Program
IORV	inadvertent open relief valve
IREP	Interim Reliability Evaluation Program
IPE	individual plant examination
ISL	interfacing system LOCA
L	late core melt with recirculation failure
LLRT	local leak rate testing
LOCA	loss-of-coolant accident
LPIS	low-pressure injection system
LPRS	low-pressure recirculation system
LWR	light-water-reactor
MW	megawatt
NCF	no containment failure
NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
NRC/RES	U.S. Nuclear Regulatory Commmission, Office of Nuclear Regulatory Research
NRR	Office of Nuclear Reactor Regulation
PORV	power operated relief valve
PRA	probabilistic risk assessment
PWR	pressurized water reactor
RCP	reactor coolant pump
RCS	reactor coolant system

NOMENCLATURE (Cont'd)

RHR	residual heat removal system
RPS	reactor protection system
RPV	reactor pressure vessel
RSS	Reactor Safety Study
RWST	refueling water storage tank
S	small LOCA
SARP	Severe Accident Research Program
SARRP	Severe Accident Risk Reduction Program
SBO	station blackout
SI	safety injection
SG	steam generator
SGTR	steam generator tube rupture
SNL	Sandia National Laboratories
STCP	Source Term Code Package
SWS	service water system
T	transient
USI	unresolved safety issue
ZPSS	Zion Probabilistic Safety Study

1. EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) has formulated an approach for a systematic safety examination of existing plants to determine whether particular severe accident vulnerabilities are present and what changes are desirable to ensure that there is no undue risk to public health and safety.

The Industry Degraded Core Rulemaking Program (IDCOR) selected four reference plants for detailed analysis: Peach Bottom, Grand Gulf, Sequoyah, and Zion. The IDCOR analyses performed for the reference plants have been documented together with the methodology used for the analyses and the technical basis supporting the methodology.

Parallel with the IDCOR work, the NRC under the Severe Accident Research Program (SARP), performed risk assessments, audit calculations, sensitivity studies, and uncertainty analyses for five plants. The five plants considered by SARP were Peach Bottom, Grand Gulf, Sequoyah, Zion and Surry.

The purpose of this effort is to review all of the IDCOR and SARP analyses performed for the reference plants, understand the reasons for the differences, and then use the experience gained from these reviews for developing guidelines and criteria that identify plant features and operator actions that were found to be important for either preventing or mitigating severe accidents in each plant type. In turn, these guidelines should prove helpful in the systematic safety examination of individual plants.

Three basic objectives or goals for this severe accident program apply equally to all plant types:

- Goal 1: Mitigate fission-product releases.
- Goal 2: Control the frequency of high-consequence sequences.
- Goal 3: Reduce high core-damage frequency.

The aim was, therefore, to develop detailed guidelines and criteria that could be used to achieve these goals during the examination of individual plants.

"Guidelines," as used in this report, identify those plant features and operator actions that were found to be important to either preventing or mitigating severe accidents in the reference plant studies. These guidelines provide a list of plant features and operator actions that the utilities can use as part of each individual plant examination (IPE). It is not the intent of this report to specify a set of improvements for either the reference plant or for any other plant which would be sufficient to achieve a certain level of safety. Instead, the guidelines indicate potential improvements in various areas of plant design and operation of which each utility should be aware when conducting its IPE and making decisions on plant improvements. The intent is to provide guidance to the analyst performing an IPE, as to the plant features and operator actions which were found to reduce overall risk. It is prudent to check whether potential improvements identified in studies of other similar plants can be of help in improving overall plant performance. The guidelines contained in this report, therefore, complement the IPEs.

"Criteria," as used in this report, are the attributes which have been identified as important to assess the performance of plant features and operator actions identified in the guidelines. The criteria provide deterministic (as opposed to probabilistic) performance measures which are judged to be helpful to implement the concepts contained in the guidelines. When a decision is made to provide an item or an action specified in a guideline, the utility should address a set of questions relating to the design, operation and availability of the needed equipment and the training of operators. For the example of containment spray guidance, the capacity of the spray system, the duration of the water source, the selection of setpoints to initiate sprays, the availability of applicable procedures and the accessibility of certain valves by operators should be addressed. The criteria on containment spray provide helpful information in assessing spray capability in each individual plant.

Based on an extensive review of prior severe accident investigations, the authors have provided a set of guidelines and associated criteria which can be used to assess the capability of individual pressurized water reactor (PWR) plants with large-volume containments, to cope with severe accidents. Although much of the work is based on probabilistic risk assessments (PRAs), the guidelines and criteria are deterministic in nature. That is the criteria describe specific features of key systems and operational procedures which have been found helpful in reducing the likelihood of severe accidents. The guidelines and criteria take into account detailed severe accident experiments and analyses performed by the NRC/RES, the nuclear power industry and foreign governments.

The following sections present the insights gained from reviewing the PRAs. Specifically, the IDCOR Zion Integrated Containment Analyses[1] and the SARP Zion reports[2-4] were reviewed in detail. These studies were compared with the original Surry risk assessment in the Reactor Safety Study (RSS) (WASH-1400)[5] and relevant PWR PRAs for other plants, including, Oconee,[6] and Surry.[7]

1.1 Core-Damage Profile

PRAs for PWRs have indicated that loss of component cooling water (CCW), transients (including station blackout) and small LOCAs tend to dominate the core-damage frequency (CDF) estimates. There was no consistent pattern of relative ranking of transient sequences across all of the studies.

For the Accident Sequence Evaluation Program (ASEP) study of Zion, accidents involving loss of CCW appeared as the dominant contributor to the CDF (about 80%). However, the quantification of this sequence appears to be very conservative since it is dominated by low pressure pipe rupture events. These events were not identified by IDCOR and therefore the CCW failure sequence is not a significant contributor in the IDCOR study.

In addition to the CCW failures, ASEP indicated that sequences involving loss of offsite power and small LOCA are the dominant core-damage sequences for Zion. Other PWR studies indicate pump seal LOCAs, loss of ac or dc buses, transients with loss of feedwater, anticipated transients without scram (ATWS) and interfacing systems LOCA (ISL) are all important contributors to PWR CDF.

1.2 Consequence Analysis

The assessment of core-meltdown phenomena and containment response in the PRAs indicated that the relatively large Zion containment is not vulnerable to overpressurization because of buildup of noncondensable gases. Both IDCOR and the NRC Severe Accident Risk Reduction Program (SARRP) indicate that early containment failure is unlikely. Thus, containment function (reducing the source term) is preserved for almost all cases.

Although the frequency of early containment failure is estimated to be low for both Zion and Surry (about 5% mean), the fission-product releases may be large and such sequences are a dominant contributor to risk in the Reactor Risk Reference Document.[8] Other large-volume containments with different reactor cavities (i.e., less susceptible to quenching of the core debris), different containment strengths (e.g., steel shell), and different volumes may be more susceptible to direct heating or hydrogen combustion. Direct containment bypass sequences (ISL) may also result in large fission-product releases.

1.3 Guidelines and Criteria

The guidelines have been developed to translate the three goals of the severe accident program into deterministic criteria for assessing each plant's response for the dominant types of core-damage sequences.

Each guideline is provided with a detailed list of criteria which provide helpful information to assess the performance of plant features and operator actions identified in the guidelines.

1.3.1 Mitigate Fission-Product Releases

For a PWR with a large-volume containment, the dominant core damage sequences were found to be loss of CCW, loss of service water, small breaks (including pump seal LOCAs and ISLs), and transients (including ATWS and station blackout (SBO)). In order to minimize off-site consequences, the containment systems must be able to fulfill their role of restricting fission-product releases even under severe accident conditions.

Guideline 1 - Maintain Containment Integrity

The most severe consequences of core-damage accidents in Zion are from accidents which result in early containment failure or bypass. The SARRP analyses indicate that direct containment heating and hydrogen combustion may cause such early failure. In order to assess the importance of such early failures in other large dry containments one guideline is provided in Table 1.1, pending further resolution of early containment failure issues.

1.3.2 Control the Frequency of High-Consequence Sequences

Guideline 2 - Interfacing Systems LOCA (Including Steam Generator Tube Rupture (SGTR))

Interfacing systems LOCA (ISL) could be a significant risk contributor because of the potentially high release even though it is usually a highly unlikely event. Although SGTR with one or more tubes ruptured bears the

characteristic of a small LOCA, it is unique in the sense that it is also a potential containment bypass LOCA, releasing reactor coolant into the secondary side of the steam generators (SGs). Thus the SGTR provides several potential paths for release of fission product to the environment outside the containment via the main steamline, turbine, turbine bypass, condenser, condenser exhaust, SG atmospheric steam dump valves or safety valves and the SG blowdown line. Thus, despite its relatively small CDF (usually a few percent of the overall CDF from internal events), Guideline 2 has been developed to prevent the occurrence of SGTR and ISL events and to mitigate the potentially high fission-product releases if one occurs.

1.3.3 Reduce High Core-Damage Frequency Sequences

The major contributors to the core damage for Zion have been identified as loss of CCW, small LOCA (S_2), and SBO. Thus, if the frequencies of these accident sequences (or a subset of them) can be reduced, then the overall CDF will be substantially reduced.

Guideline 3 - Component Cooling Water (CCW)

An event initiated by loss of CCW may lead to reactor coolant pump (RCP) seal LOCA sequences because of loss of RCP seal cooling, under the condition that high pressure injection fails because pumps in the high pressure injection systems are also cooled by CCW. The contribution of this initiator to CDF can be reduced by r cing the frequency of loss of the CCW system. Specific criteria have been provided to ensure that loss of CCW does not lead to an unacceptably high CDF.

Guideline 4 - Operator Response for Recirculation

After CCW failures, the small LOCA initiator (S_2) is the largest contributor to core damage at Zion. Major contributors to the S_2 sequences are operator failures in the recirculation switchover operation and common cause failures of the containment sump suction valves and the high pressure injection pump suction valves. Although the sequences involving loss of containment heat removal (CHR) were found only marginally important for Zion, the main contributor to these sequences is also the failure in the recirculation switchover.

Guideline 5 - Station Blackout

In Zion, station blackout (SBO) sequences are important contributors to CDF. The accident sequences in SBO are characterized by two categories of postulated events: First, ac power is not recovered before battery depletion which, in turn, defeats the turbine-driven auxiliary feedwater system (AFWS) pump. Second, SBO causes a RCP seal LOCA because of failures of seal injection flow and RCP thermal barriers resulting from loss of the CCW and service water systems.

The AFWS is the normal means of decay heat removal in small LOCA and transient events, including a normal plant shutdown. The Zion PRAs indicate that the core-damage sequences involving failure of the AFWS are related to (1) SBO which is dominated by failure of the AFWS turbine-driven pump because of dc bus failure or battery depletion, and (2) the loss of a dc bus initiator

followed by failures of the AFWS motor-driven pumps and by failure of the RCS feed & bleed cooling. The criteria under Guideline 5 address the availability of an AFWS turbine-driven pump during station blackout and the operator actions required to mitigate a loss-of-AFWS event if it occurs.

For accidents involving the loss-of-offsite power and onsite emergency power, the NRC recommends examining the proposed SBO rule for applicability. The criteria associated with Guideline 5 are intended to emphasize the need to search for plant specific features and potential common cause failures which could disable systems required to work during an SBO. For individual plants which are found to have a vulnerability to SBO, the criteria highlight the importance of proper emergency procedures and operator training in recovering from an SBO event.

Guideline 6 - Reactor Coolant System (RCS) Feed & Bleed Cooling

One potentially effective means of making up core inventory in case the high-pressure injection systems are unavailable, either in injection or in recirculation phase, is to depressurize the RCS by the power operated relief valves (PORVs). After depressurization, the low-pressure injection system (LPIS) can be actuated for the makeup operation. This emergency procedure appears to have a significant impact on reducing the core-damage frequencies. To attain success in this procedure, the operator must open the PORVs and have at least one of the LPIS trains available for RCS makeup.

Guidelines 7 and 8 - Low-Pressure Injection and RCS Depressurization by Secondary Blowdown

A second potentially effective means of making up core inventory in case the high pressure systems are unavailable, either in injection or in recirculation phase, is via secondary blowdown, that is to depressurize the RCS by heat removal through the SGs. After sufficient heat removal, the LPIS can be actuated for the RCS makeup operation. This action appears to have a significant impact on reducing the core damage frequencies. To attain success, the operator must open the SG atmospheric steam dump valves, maintain AFW or main feedwater to the SG and have one of the LPIS trains available for RCS makeup.

Guideline 9 - Anticipated Transients Without Scram

Anticipated transients without scram (ATWS) is not a significant contributor to core damage at Zion. However, several PWR PRAs have found ATWS to be an important event to consider. One factor in the low CDF estimates for ATWS is the credit given to the manual scram and the emergency boration using the charging system to deliver borated water to the reactor vessel. Failures of the manual scram and the emergency boration are dominated by failures of operator actions.

Guideline 10 - Support System Interdependencies

Although the importance is difficult to quantify, one of the insights of most risk assessment studies is the importance of support system interdependencies. For example, a draft of the SARRP Peach Bottom study indicated that loss of all service water was a dominant contributor to core melt. The recent revision to the sequence studies have reduced it to one percent of the overall

core melt. In order to ensure that support system vulnerabilities do not cause an unacceptably high CDF for other PWR large-volume containment plants, the authors have developed this guideline to help assess any weaknesses of the support systems.

Guideline 11 - Flooding of Emergency Equipment

Internal floods have been found to be a significant contributor to core damage in only one PWR plant. However, the concerns appear to be of general applicability to other designs. Thus, this guideline was developed for all PWR plants to to assess the potential for flooding of safety-related equipment.

1.4 Using the Guidelines and Criteria

Numerous investigations, including PRAs, have been performed for the reference plants and similar plants by both the NRC and the nuclear power industry. The insights gained from many of the studies have been used in developing the guidelines and criteria contained in this report (including the other volumes of this report). These guidelines and criteria are issued to provide guidance to the analyst performing an IPE. This guidance is in the form of plant features, operator actions and the criteria for assessing those features and actions found to be helpful in reducing the overall risk for Zion and other PWR plants. Thus, the guidance is given to provide a resource in examining the subject plant to determine if the same, or similar, guidelines will be of value in reducing overall plant risk. These guidelines and criteria are intended to be used solely as guidance.

1.5 References for Section 1

1. "Zion Nuclear Generating Station and Integrated Containment Analysis," IDCOR Technical Report T.23.1Z, March 1985.

2. T. Wheeler, "Analysis of Core Damage Frequency from Internal Events: Zion Unit 1," Sandia National Laboratories, NUREG/CR-4550, Volume 7, October 1986.

3. M. L. Corradini et al., A Review of the Severe Accident Risk Reduction Program (SARRP) Containment Event Trees," NUREG/CR-4559, U.S. Nuclear Regulatory Commission, May 1986.

4. M. Khatib-Rahbar et al., "Evaluation of Severe Accident Risks and Potential for Risk Reduction: Zion Power Plant," Brookhaven National Laboratories, Volume 5, Draft Report for Comment, February 1987.

5. "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, NUREG/75-014, October 1975.

6. W. R. Sugnet et al., "Oconee PRA - A Probabilistic Risk Assessment of Oconee Unit 3," Duke Power Co., NSAC-60, June 1984.

7. R. C. Bertucio et al., "Analysis of Core Damage Frequency from Internal Events: Surry, Unit 1," Sandia National Laboratories, NUREG/CR-4550, November 1986.

8. "Reactor Risk Reference Document," U.S. Nuclear Regulatory Commission, NUREG-1150, Draft for Comment, February 1987.

Table 1.1 Guidelines for Preventing and Mitigating Severe Accidents in a PWR with a Large-Volume Containment

Guideline		Description
		Mitigate Fission-Product Releases:
1		Maintain Containment Integrity
	1.A.	Provide Containment Spray and Long Term Heat Removal
	1.B.	Provide Filtered Venting
	1.C.	Assessment of Early Containment Failure
		Control the Frequency of High-Consequence Sequences:
2		Interfacing Systems LOCA (Including Steam Generator Tube Rupture)
	2.A.	Prevent Overpressurization of Low Pressure Systems
	2.B.	Prevent Steam Generator Tube Rupture and Minimize Its Consequences
		Reduce High Core-Damage Frequency Sequences:
3		Component Cooling Water
	3.A.	Provide Adequate Cooling to Engineered Safety Features Systems
4		Operator Response for Recirculation
	4.A.	Provide Adequate Core Recirculation Cooling
	4.B.	Provide Adequate Containment Heat Removal
5		Station Blackout
	5.A.	Provide Operator Response During Station Blackout
6		Reactor Coolant System (RCS) Feed & Bleed Cooling
	6.A.	Provide Operator Response for RCS Feed & Bleed Cooling
7		Low-Pressure Injection
	7.A.	Improve Availability of Low Pressure Injection
8		Reactor Coolant System (RCS) Depressurization by Secondary Blowdown
	8.A.	Provide RCS Depressurization Capability
9		Anticipated Transients Without Scram (ATWS)
	9.A.	Provide Operator Response During ATWS
10		Support System Interdependencies
	10.A.	Examine Support System Interdependencies
11		Flooding of Emergency Equipment
	11.A.	Prevent or Mitigate Internal Flooding

2. INTRODUCTION

2.1 Background

The U.S. Nuclear Regulatory Commission (NRC) has formulated an approach for a systematic safety examination of existing plants to determine whether particular severe accident vulnerabilities are present and what changes are desirable to ensure that there is no undue risk to public health and safety.

The Industry Degraded Core Rulemaking Program (IDCOR) selected four reference plants for detailed analysis, namely:

- Peach Bottom (a BWR with a Mark I containment)
- Grand Gulf (a BWR with a Mark III containment)
- Zion (a PWR with a large dry containment)
- Sequoyah (a PWR with an ice condenser containment)

The IDCOR analyses performed for the above reference plants have been documented together with the methodology used for the analyses and the technical basis supporting the methodology.

Parallel with the IDCOR work, the NRC under the Severe Accident Research Program (SARP), performed risk assessments, audit calculations, sensitivity studies, and uncertainty analyses for five plants. The five plants considered include the above four IDCOR reference plants, and, in addition

- Surry (a PWR with a subatmospheric containment)

The purpose of this effort is to review all of the IDCOR and SARP analyses performed for the reference plants, understand the reasons for the differences, and then use the experience gained from these reviews for developing guidelines and criteria that identify plant features and operator actions found to be important for either preventing or mitigating severe accidents in each plant type. In turn, these guidelines should be helpful in the systematic safety examination of individual plants.

The IDCOR Zion analysis[1] was documented in March 1985 and was supplemented by additional sensitivity studies in July 1985. The SARP Zion reports[2-4] were reviewed in draft form during 1986. These reports were published early in 1987 and were summarized in the "Reactor Risk Reference Document" (NUREG-1150),[5] which was published for comment in February 1987. The experience gained from the review of these Zion studies along with other PWR PRA studies (namely, Surry, Oconee and Millstone III) was used to generate the guidelines and criteria which are the subject of this report.

2.2 Objectives

Three basic objectives or goals for this severe accident program apply equally to all plant types:

- Goal 1: Mitigate fission-product releases.
- Goal 2: Control the frequency of high-consequence sequences.
- Goal 3: Reduce high core-damage frequency.

The aim was, therefore, to develop detailed guidelines and criteria that could be used to achieve these goals during the examination of individual plants. The guidelines and criteria are defined in the sections that follow.

2.2.1 Guidelines

"Guidelines," as used in this report, identify those plant features and operator actions that were found to be important to either preventing or mitigating severe accidents in the reference plant studies. These guidelines provide a list of plant features and operator actions that the utilities can use as part of each individual plant examination (IPE). It is not the intent of this report to specify a set of improvements for either the reference plant or for any other plant which would be sufficient to achieve a certain level of safety. Instead, the guidelines indicate potential improvements in various areas of plant design and operation of which each utility should be aware when conducting its IPE and making decisions on plant improvements. The intent is to provide guidance to the analyst performing an IPE, as to the plant features and operator actions which were found to reduce overall risk. It is prudent to check whether potential improvements identified in studies of other similar plants can be of help in improving overall plant performance. The guidelines contained in this report, therefore, complement the IPEs.

The three objectives or goals were noted as applying equally to all plant types. Although the goals are independent of plant type, the guidelines that are needed to achieve the goals are plant dependent. In general terms, Goal 1 implies that there should be effective means of mitigating the fission-product releases for the broad classes of accident sequences which dominate the core-damage frequency. Therefore, these dominant accident sequences have to be determined and those plant features and operator actions that are available to mitigate the release of fission products have to be identified. Only then can detailed guidelines be developed to ensure that these dominant accident sequences can be mitigated.

There may be accident sequences for which a specific plant will have substantial fission-product releases (e.g., containment bypass sequences). Thus, for such sequences Goal 1 may be difficult to achieve. Therefore, all reasonable steps are identified which could reduce the frequency of these potentially high consequence sequences (namely Goal 2). Again, the accident sequences have to be identified and plant vulnerabilities and/or operator actions that lead to core damage for these sequences also have to be identified. Detailed guidelines can then be developed which will aid in assessing an individual plant's capability to prevent these sequences from occurring.

It is also desirable to ensure that the overall core-damage frequency is low (namely Goal 3). Again, the dominant accident sequences have to be found so that detailed guidelines can be developed to reduce the frequency of these sequences, if necessary.

In general, the following screening process was used to determine whether or not to develop a particular guideline:

- any accident sequence with a core-damage frequency greater than 10^{-6} per reactor year.

- any sequence that contributed to more than 5% of the total core-damage frequency

- any event that caused a conditional probability of early containment failure greater than 0.1

- any sequence that resulted in containment bypass with a frequency greater than 10^{-7} per reactor year

- any sequence that was judged to be uniquely important (example, very severe consequences)

This screening process led to the development of guidelines that can be used in the systematic safety examination of other PWRs with large-volume containments. For example, the guideline for containment spray and long-term heat removal was identified as an item that would help to achieve Goal 1 (namely, to mitigate fission-product releases) for the PWR large-volume reference plant. Therefore, in the safety examination of other PWRs with large-volume containments, the capability for containment spray and long-term heat removal may need to be carefully assessed.

The development of a particular guideline for the PWR large-volume reference plant does not imply that this plant or any of the other plants in this category need to conform to this guideline. It simply means that analyses have indicated that this particular guideline has the potential to significantly reduce risk. Thus, the guidance is given to provide a resource in examining the subject plant to determine whether the same or similar guidelines will be of value in reducing overall plant risk. Whether or not the guideline is useful or needed in a particular PWR with a large-volume containment depends on plant-specific details and is beyond the scope of this report and is therefore not addressed here.

2.2.2 Criteria

"Criteria," as used in this report, are the attributes which have been identified as important to assess the performance of plant features and operator actions identified in the guidelines. The criteria provide deterministic (as opposed to probabilistic) performance measures which are judged to be helpful to implement the concepts contained in the guidelines. When a decision is made to provide an item or an action specified in a guideline, the utility should address a set of questions relating to the design, operation and availability of the needed equipment and the training of operators. For the example of containment spray guidance, the capacity of the spray system, the duration of the water source, the selection of setpoints to initiate sprays, the availability of applicable procedures and the accessibility of certain valves by operators should be addressed. The criteria on containment sprays provide helpful information in assessing spray capability in each individual plant.

The criteria address the general issues of (1) survivability of equipment (i.e., whenever credit is given for a system or a component to mitigate the accident, the ability of the equipment to function under environmental and fluid dynamic loads associated with severe-accident sequences must be taken into account), (2) equipment capabilities, capacities, and duration of

operability, (3) accessibility of equipment, (4) availability of support systems, (5) identification of necessary components, (6) identification of important operator actions, and (7) identification of parameters for initiation of mitigating systems and operator actions.

2.3 Organization of the Report

This report describes detailed guidelines and criteria for preventing and mitigating severe accidents in PWRs that have a large-volume containment. It is the fourth volume in a series of five, that deal with guidelines and criteria for several different reactor and containment types. Other volumes in the series are:

- Volume 1: BWRs with Mark I Containments
- Volume 2: BWRs with Mark II Containments
- Volume 3: BWRs with Mark III Containments
- Volume 5: PWRs with Ice Condenser Containments.

Appendix A of this volume contains a review of the IDCOR and SARP analyses for a PWR with a large-volume containment along with other pertinent studies. The insights gained from these studies lead to the identification of the strengths and vulnerabilities of a PWR with a large-volume containment. In Section 3, the three basic goals of the program are related to the relevant design features and operating characteristics of a PWR with a large-volume containment. The guidelines recommended to achieve the three goals are therefore initially developed in Section 3. In Section 4, the guidelines are restated and detailed criteria are developed for each guideline.

2.4 References for Section 2

1. "Zion Nuclear Generating Station and Integrated Containment Analyses," IDCOR Technical Report T23.1Z, March 1985.

2. T. Wheeler, "Analysis of Core Damage Frequency from Internal Events: Zion,Unit 1," Sandia National Laboratories, NUREG/CR-4550, Volume 7, October 1986.

3. M. L. Corradini et al., "A Review of the Severe Accident Risk Reduction Program (SARRP) Containment Event Trees," U.S. Nuclear Regulatory Commission, NUREG/CR-4569, May 1986.

4. M. Khatib-Rahbar et al., "Evaluation of Severe Accident Risks and Potential for Risk Reduction: Zion Power Plant," Brookhaven National Laboratories, NUREG/CR-4551, Vol. 5, Draft Report for Comment, February 1987.

5. "Reactor Risk Reference Document," U.S. Nuclear Regulatory Commission, NUREG-1150, Draft for Comment, February 1987.

3. DEFINITION OF GOALS AND RELEVANT FEATURES OF PWRS WITH LARGE-VOLUME CONTAINMENTS

In Section 2 of this report, the concept of three basic objectives or goals for this severe accident program was introduced. The concept applied equally to all plant types. In this section, the three goals are related to the relevant design features and operating characteristics of a PWR with a large-volume containment for accident sequences and containment failure modes found to be important in Appendix A. This includes consideration of both favorable and unfavorable severe accident attributes. Table 3.1 summarizes the dominant accident sequences identified in the ASEP study for Zion. PRAs for other PWRs have been reviewed for the possibility of other unique sequences which could contribute to the core-damage frequency.

Screening criteria have been used to identify those sequences which need to be addressed by severe accident guidelines for each goal. Specifically:

- For Goal 1 (Mitigate fission-product releases), all sequences have been examined which represent 5% of the core-melt frequency or are estimated to occur more often than 10^{-6} per reactor year and which result in a conditional probability of early containment failure greater than 0.1.
- For Goal 2 (Control the frequency of high-consequence sequences), all sequences have been examined which result in containment bypass and are estimated to occur more often that 10^{-7} per reactor year.
- For Goal 3 (Reduce high core-damage frequency sequences), all sequences have been examined which "have the potential to occur" more frequently than 10^{-6} per reactor year. Note that this screening criterion has been used to identify potential vulnerabilities from risk assessment insights which do not necessarily apply to Zion itself, but may apply to other PWR large-volume plants.

This section provides the link between the goals (developed in Section 2) and the guidelines (developed in Section 4) that may be used to assess the capability of specific plants to meet these goals. This section is organized into three subsections, which correspond to the three goals.

3.1 Mitigate Fission-Product Releases

Goal 1 requires that there shall be effective means of mitigating the fission-product releases for the broad classes of accident sequences that may lead to core damage in a PWR with a large-volume containment. In Appendix A the most important contributors to the core-damage frequency (CDF) were found to be small break loss-of-coolant accidents (LOCAs) (including pump seal LOCAs and the interfacing systems LOCA sequence), large break LOCA, transients with feedwater failure (including station blackout and loss of dc or ac buses) and loss-of-component cooling water or service water. No specific accident sequences for which mitigation by the large-volume containment is ineffective have been identified for either Surry or Zion. Rather, three generalized threats to the containment (direct heating, hydrogen combustion and containment heat removal (CHR) failure) have been treated in the SARRP uncertainty assessment. These three phenomena appear to dominate containment failure and risk for PWRs with large-volume containments. Specifically, the combined

effects of direct heating and hydrogen combustion pose the only significant threat to early containment failure while CHR failure poses a significant threat to long term containment integrity. This section concentrates on these broad threats to containment integrity for which plant features provide significant mitigating fission-product release.

3.1.1 Plant Vulnerabilities

The Zion containment is a large-volume containment design. The volume of the containment makes it effective against pressure buildup because of noncondensable gas generation and hydrogen combustion during a core-meltdown accident. However, even a large-volume containment will fail, eventually, unless containment heat removal (CHR) is preserved. There are differences between the IDCOR[1,2] and ASEP/SARRP[3-5] analyses as to how long it will take to pressurize a large-volume containment to its ultimate capacity after the core debris has failed the reactor pressure vessel (RPV) and the reactor cavity has dried out; but, both studies concluded that for some cases the containment will eventually fail. Therefore, unless CHR is preserved, the containment will fail for some accidents because of overpressure. If containment failure occurs, a significant inventory of fission products in the containment atmosphere could be released to the environment.

An inspection of the Zion containment configuration indicates that the cavity below the RPV would tend to confine the core debris after a core meltdown accident. For many sequences the water from the emergency core cooling (ECC) accumulators and containment sprays will flood the lower cavity, cooling the core debris. However, for some sequences the containment sprays may not be initiated. Without containment sprays the water in the reactor cavity would be limited. There are differences between the IDCOR[2] and Battelle Columbus Laboratories (BCL)[4] analyses as to how hot the core debris will be during core-concrete interactions and as to how much of the less volatile fission products will be released. However, at this time the possibility of the core debris remaining hot or eventually drying out and releasing significant quantities of fission products has not been ruled out.

The most important vulnerability of the large-volume containment appears to be the possibility of early containment failure, because of hydrogen deflagrations or detonations and direct heating. If early containment failure does not occur there is still a significant probability of late containment failure (about 6%), because of buildup of noncondensable gases or steam after failure of CHR.

3.1.2 Mitigating Features

For early containment failure sequences the containment sprays provide an effective device for removing any fission-product aerosols that might pass through it. The sprays also help flood the reactor cavity and reduce the likelihood of direct heating.

Long term overpressure failure of the containment can be prevented by intensive cooling of ECC water and of recirculating spray water as well as operation of the fan cooler system.

The most severe consequences of core-melt accidents in Zion are from accidents with early containment failure. The SARRP uncertainty analysis[5] indicates that hydrogen detonations or deflagrations may contribute to such early failure. In order to prevent hydrogen detonations and deflagrations for some vulnerable plants, hydrogen igniters may be necessary. The NRC has identified the threat of hydrogen to large-volume containments as a generic issue (GI-121).

3.1.3 Maintain Containment Integrity

The above discussion has identified several features of the PWR plant with a large-volume containment that have the potential to help achieve Goal 1, namely, mitigating fission-product releases. Therefore, a guideline has been developed and related to those features which will aid in assessing whether specific plants can meet Goal 1. The guideline addresses the following three areas:

- containment sprays and long-term heat removal,
- filtered venting, and
- assessment of early containment failure potential.

3.2 Control the Frequency of High-Consequence Sequences

The plant features identified in Section 3.1 can effectively mitigate fission-product release for the broad classes of accident sequences that were found to dominate the core-damage frequency (CDF). However, one accident sequence was identified in Appendix A for which substantial reducing of fission-product release for the PWR large-volume containment plant cannot be assured. The interfacing systems LOCA sequences appear to meet the screening criteria ($>10^{-7}$/reactor year) for Zion, but its importance in other PRAs indicate that guidance should be developed to ensure that specific plant vulnerabilities do not make them more important contributors to risk for other PWR large-volume containment plants.

3.2.1 Interfacing Systems LOCA (Including Steam Generator Tube Rupture (SGTR))

The interfacing systems LOCA (ISL) would open up a direct path from the reactor coolant system to the auxiliary building thus bypassing the containment. The only plant feature pertinent to mitigating this event is the auxiliary building, which may not be sufficient on its own to ensure low fission-product releases to the environment. Therefore, the frequencies of accident sequences resulting from this initiator should be controlled (Goal 2). In Section 4.2, a guideline and associated criteria are developed to try to meet Goal 2. It is noted that BNL is currently performing a study on ISL to provide technical support to the Reactor Safety Issues Branch of NRC for a meaningful resolution of this generic issue.

The steam generator tube rupture (SGTR) event would provide an open pathway for early radioactive release to the atmosphere if the core damage is not arrested in a timely fashion and the release paths in the secondary side (e.g., the main steam isolation valve of the ruptured steam generator (SG)), and the associated SG atmospheric steam dump valves are not isolated properly. Although this initiator has not been identified as a dominating

contributor to the overall CDF for Zion, it has been identified for other PWRs and it could be important to risk because it can lead to accident sequences which bypass the containment. Thus, the frequencies of these sequences should also be controlled (Goal 2). In Section 4.2, a guideline and associated criteria are developed to ensure that other PWRs with large-volume containments do not have high risk because of containment bypass events.

3.3 Reduce High Core-Damage Frequency Sequences

The set of accident sequences which were identified in the IDCOR and SARP studies as dominant contributors to CDF for PWR with large-volume containments are presented in Section A.1. This provides the basis for the assessing the general types of sequences which can be expected to be important for other PWRs with large-volume containments.

3.3.1 Component Cooling Water (CCW)

The most important contributors to the core damage at Zion (excluding external events) were found to be loss of CCW, LOCAs and station blackout sequences.

Loss of CCW, as an initiator, may cause failures of seal injection flow and failure of the thermal barrier and may lead to reactor coolant pump (RCP) seal LOCA sequences because of loss of RCP seal cooling. Note also, that, depending on the redundancy and heat loads of the CCW system and the essential service water (ESW) system in a particular plant, the loss of ESW could be more "critical" than the loss of CCW with regard to RCP seal LOCA, depending on whether the charging pumps (which provide RCP seal injection water) are cooled by the CCW or by the ESW system. Even for RCPs which are cooled by CCW, an RCP seal LOCA may result from the loss of the ESW since the loss of ESW, if it is not rapidly recovered, would lead to degradation of CCW. Reliability of the CCW including the piping integrity should be assessed to determine the likelihood of CDF resulting from this initiator for all PWRs.

3.3.2 Operator Response for Recirculation

Major contributors to the LOCA sequences are related to the high-pressure recirculation switchover which involves several operator failures and the common cause failures of the containment sump suction valves and high-pressure injection pump suction valves. This type of sequence has been found in other PWR PRAs even though emergency operating procedures of the plants usually address this recirculation switchover operation.

3.3.3 Station Blackout

Characteristics of the important accident sequences in station blackout (SBO) are battery depletion defeating the turbine-driven AFWS pump, and RCP seal LOCA because of failures of seal injection flow and RCP thermal barriers resulting from loss of the CCW and service water systems. SBO is currently the subject of an unresolved safety issue (USI A-44). Thus, development of guidelines and criteria for SBO are considered to be appropriate pending resolution of USI A-44. For individual plants which are found to have a vulnerability to SBO, the criteria developed in Section 4.3 highlight the importance of emergency procedures and operator training in recovering from an SBO event.

3.3.4 Reactor Coolant System Feed and Bleed Cooling

The success of reactor coolant system (RCS) feed and bleed cooling (high-pressure injection (HPI) and steam relief through the RCS power operated relief valves (PORVs)) has a significant impact on the estimated CDF for many PWRs. However, the capacity of the PORVs and the HPI varies from plant to plant. Thus, a guideline and associated criteria are provided to assess RCS feed and bleed capacity and to ensure that the operator training and emergency procedures are sufficient to achieve successful feed and bleed cooling. Note that feed and bleed cooling is a decay heat removal issue (USI A-45).

3.3.5 Low-Pressure Injection

Failure of low pressure injection in a large LOCA is a significant contributor to the core damage at Zion. Improving availability of the low pressure injection system (LPIS) would not only reduce the frequency of this sequence but also enhance the RCS depressurization by secondary blowdown. Depressurization and LPIS operation are potentially effective means of making up core inventory in case the HPI systems are unavailable.

3.3.6 Reactor Coolant System (RCS) Depressurization

Small break LOCAs with failure of the high-pressure injection system (HPIS) have been found to be a contributor to CDF for many PWR risk assessments. Changes to emergency procedures to allow RCS depressurization by secondary blowdown are being reviewed by nuclear industry and the NRC. Although the net risk reduction has yet to be assessed it is considered prudent to provide a guideline to assess this capability at each plant.

3.3.7 Anticipated Transients Without Scram (ATWS)

Another set of accident sequences that may defeat several plant safety features and that contribute to the core damage is related to an ATWS. Although ATWS does not appear to contribute to the CDF for Zion, it is a dominant contributor for other PWRs including Surry and Millstone-3. The ATWS initiated by loss of main feedwater was fount to be most severe. Failure of secondary cooling via the SG and RCS PORV relief in this sequence could lead to a small LOCA and damage the check valves between the RCS system and the HPI systems which may lead to core damage. In particular, if the turbine fails to trip in this sequence, its demand for steam on the SGs will cause them to dry out rapidly, rendering the secondary cooling by the AFWS ineffective.

3.3.8 Support System Interdependencies

Most PRAs have stressed the importance of unrecognized interdependencies having the potential to compromise the performance of many critical safety systems. In many cases risk assessment studies have identified such vulnerabilities very early in the study and fixes have been made which substantially reduced risk.

The ASEP study of Zion indicated a vulnerability to loss of CCW. PRAs for other PWRs have also found vulnerabilities to service water and ac and dc busses. Thus, it is extremely important that each plant attempt to identify any unique support system vulnerabilities.

3.3.9 Flooding of Emergency Equipment

As excessive water release into a portion of the turbine building or auxiliary building outside the primary containment which houses a concentration of safety-related equipment raises the possibility of a common-mode event disabling key equipment. At least one plant with a large-volume containment has been identified in which the location of safety equipment makes the internal flooding initiator a substantial contributor to CDF.

To help ensure that other PWR plants which may have a similar safety-related equipment flooding potential can be identified, a guideline and associated criteria are provided in Section 4.3 which may be used to screen for such vulnerabilities.

3.4 References for Section 3

1. "Risk Reduction Potential," Energy Incorporated, IDCOR Technical Report 21.1, June 1985.

2. "Zion Nuclear Generating Station - Integrated Containment Analysis," IDCOR Technical Report T23.1Z, March 1985.

3. J. A. Gieseke et al., "Radionuclide Release Under Specific LWR Accident Conditions," BMI2104, Volume VI, PWR-Large, Dry Containment Design (Zion Plant), Battelle Columbus Laboratories, July 1984.

4. R. S. Denning et al., "Report on Radionuclde Release Calculations for Selected Severe Accident Scenarios to U.S. NuclearRegulatory Commission," Battelle Columbus Laboratories, NUREG/CR-4624, BMI-2139, Vol. 5, July 1986.

5. M. Khatib-Rahbar et al., "Evaluation of Severe Accident Risks and Potential for Risk Reduction: Zion Power Plants," Brookhaven National Laboratory, NUREG/CR-4551, Vol. 5, Draft Report for Comment, February 1987.

Table 3.1 Dominant Accident Sequences for Zion in the ASEP Study (Mean Values per Reactor Year)*

No.	Sequence	Core-Damage Frequency
1	CCW failure, causing failure of all charging and SI pumps, seal LOCA	5.5×10^{-6}**
2	Small LOCA, failure of recirculation cooling	1.6×10^{-5}
3	Large LOCA, failure of recirculation cooling	4.9×10^{-6}
4	Medium LOCA, failure of recirculation cooling	4.9×10^{-6}
5	Loss of offsite power, failure of AFWS, failure of feed and bleed, failure to restore ac power in one hour (recovery by four hours)	2.1×10^{-6}
6	Large LOCA, failure of low pressure injection	1.4×10^{-6}
7	Loss of offsite power, failure of AFWS, failure of feed and bleed, failure to restore ac power in four hours (recovery by eight hours)	4.6×10^{-7}
8	Loss of offsite power, CCW/SWS loss, failure to restore ac in one hour (recovery by four hours)	3.2×10^{-7}
9	Same as sequence 8, only this represents the SWS common-mode portion of the rebaselined Zion Review sequence no. no. 3 of Table A.4	3.0×10^{-7}
10	Loss of offsite power, CCW/SWS loss, failure to restore ac power in eight hours, failure of containment sprays and fan coolers	2.0×10^{-7}
11	Loss of offsite power, CCW/SWS loss, failure to restore ac power in four hours (recovery by eight hours)	1.5×10^{-7}
12	Loss of offsite power, failure of SWS, failure to restore ac power in eight hours. This sequence represents the SWS portions of the rebaselined Zion Review sequences no. 4 and no.6 of Table A.4	1.5×10^{-7}
13	Same as sequence 12 above, only this is the CCW portion of the rebaselined Zion Review sequence no. 4 of Table A.4	1.0×10^{-7}
14	Interfacing systems LOCA	1.0×10^{-7}
15	Failure of dc Bus 112, causing loss of one PORV and loss of ac Bus 148, failure of AFW	5.0×10^{-8}
Total		5.5×10^{-5}**

*From information contained in the draft ASEP report, July 1986 (Ref. 5 in Section 2).

**Revised in NUREG/CR-4551, February 1987.

4. GUIDELINES AND CRITERIA FOR PWRs WITH LARGE-VOLUME OR SUBATMOSPHERIC CONTAINMENTS

In Section 3, those accident sequences that dominate the core-damage frequency (CDF) were identified as were those that are potentially of high consequence. Vulnerabilities of large-volume containments to severe accident containment loads were discussed and those features of a PWR with a large-volume containment which are important for preventing core damage and are available for mitigating fission-product release to the environment were identified.

Based on the "insights" from the IDCOR, Zion review, and SARP studies for Zion and previous PRA studies, the following sections provide guidelines defining "deterministic, plant-specific guidance on the design features and operating characteristics which are to be examined by the utilities,"[1] and criteria defining "deterministic standards for judging the acceptability of plant features."[1] From SECY-86-76[2] further guidance is provided in defining guidelines and criteria. These guidelines "will specify the plant features and operator actions which are considered important to ensuring acceptable risk for the reference plant."[2] Further acceptance criteria (for the various guidelines) "will specify the attributes necessary to ensure acceptable performance."[2]

Based on this work, eleven guidelines were developed which reflect the importance of these features to plant risk. As discussed in Section 2.2.1 these guidelines indicate areas of potential improvements for various areas of plant design and operation of which utilities should be aware when conducting assessments. It is further noted that a number of the guidelines appear to overlap various generic issues as defined by the NRC. Final resolution and disposition of these generic issues may encompass NRC-imposed requirements. However, the guidelines and criteria presented herein are intended only for the purposes noted above. The guidelines are summarized in Table 1.1.

Guideline 1 was developed to ensure the capability to mitigate fission-product releases (Goal 1) with reference to maintaining containment integrity.

Guideline 2 was developed for controlling the frequency of high-consequence sequences (Goal 2) with reference to minimizing interfacing systems LOCA frequency.

Finally, Guidelines 3 through 11 were developed for reducing high core-damage frequency sequences (Goal 3) with reference to mitigating station blackout (SBO) sequences, mitigating anticipated transients without scram (ATWS) sequences, mitigating loss of containment heat removal sequences, enhancing reactor coolant system (RCS) depressurization, examining support system interdependencies, and mitigating internal floods.

The remainder of this section is organized into three subsections corresponding to the three basic goals. In each subsection, the corresponding guidelines are discussed from which detailed criteria are developed in order to provide standards by which each plant will be measured for compliance with the guidelines. The criteria address (see Section 2.2.2), under severe accident conditions, the general issues of (1) survivability of equipment (i.e., whenever credit is given for a system or a component to mitigate an accident, the ability of the equipment to function under the environmental conditions

and fluid dynamic loads associated with severe accident sequences must be taken into account), (2) equipment capabilities, capacities, and duration of operability, (3) accessibility of equipment, (4) availability of support systems, (5) identification of necessary components, (6) identification of important operator actions, and (7) identification of parameters for initiation of mitigating systems and operator actions.

4.1 Mitigating Fission-Product Releases

For a PWR with large-volume containment, the dominant core-damage sequences were found to be loss of component cooling water, small break LOCAs (including interfacing systems LOCA) and transients (including SBO and ATWS). In order to minimize offsite consequences for the sequences, the containment systems (both primary and secondary) should be able to retain a substantial fraction of fission products released even under these severe accident conditions. In this section, one guideline is provided with associated criteria which should help other PWRs assess the capability of their containments and thereby mitigate the release of fission products during a severe accident.

4.1.1 Maintain Containment Integrity (Guideline 1)

The phenomena that determine the magnitude of the containment pressure rise because of "direct heating" (transfer of sensible heat and oxidation energy from an aerosolized melt directly to the containment atmosphere) and hydrogen combustion are highly uncertain. However, the SARRP uncertainty analyses indicate these phenomena dominate the likelihood of early containment failure in the Zion and Surry containments. The present NRC research results indicate that direct heating is not risk dominant for Zion but its importance for other large volume containments has yet to be demonstrated.

Because this issue is not resolved, a guideline and associated criteria have been provided in Table 4.1 based on the engineering judgement that it may be important to risk in some PWRs.

4.2 Control the Frequency of High-Consequence Sequences

Accident sequences have been identified for which the PWR large-volume containment has limited means of mitigating fission-product releases. In this section, a guideline and associated criteria for controlling these potentially high-consequence sequences have been developed.

4.2.1 Interfacing Systems LOCA (Including Steam Generator Tube Rupture) (Guideline 2)

As discussed in Section 3.2, although the interfacing systems LOCA (ISL) is usually a highly unlikely event, it could be a significant risk contributor because of the potentially high fission-product releases to the environment. The objective of this guideline and the associated criteria is to ensure that the frequency of ISL events is kept at a low level.

Brookhaven National Laboratory (BNL) is presently performing a study to provide technical support to the NRC for the meaningful resolution of the generic issue related to ISL (GI-105). The criteria for this guideline should be considered to be appropriate pending resolution of the generic issue.

For the ISL guideline, the performance of equipment, systems, and operators should be assessed against specific performance criteria to ensure successful prevention of interfacing systems LOCA. The criteria relate to equipment (low-pressure systems interfacing with high-pressure systems) and operator performance (isolation and relief valve maintenance and surveillance).

Although steam generator tube rupture (SGTR) bears the characteristics of a small LOCA, it is unique in the sense that it is also a potential containment bypass LOCA, releasing primary reactor coolant into the secondary-side of steam generators, which provide several potential paths for release of fission products to the environment outside the containment via the main steam-line, turbine, turbine bypass, condenser, condenser exhaust, steam generator atmospheric relief or safety valves, and the steam generator blowdown line. Thus, despite its small contribution to risk,[3] it is important to provide the necessary measures to prevent the occurrence of SGTR and to isolate it if it occurs.

Prevention of SGTR should be aimed at removing all conceivable causes that may contribute to degradation and failure of steam generator tubes, with special emphasis placed on preventing chemical corrosion of the tubes by improved secondary side water chemistry, and/or protecting the integrity of steam generator tubes from the impact of foreign objects through improved surveillance and inspection. The primary objectives of recovery actions and mitigating the consequences of SGTR should be (1) to stop the primary-to-secondary leakage, (2) to restore reactor coolant inventory to ensure adequate core cooling, (3) to minimize the release of fission products to the surrounding environment from the ruptured steam generators, and (4) to stabilize the reactor system to regain plant control.

Several important operator actions are required for successful achievement of these objectives, including early diagnosis of SGTR, identification and isolation of the faulty steam generator (SG), manual depressurization of the RCS to stop the leakage flow and prevention of main steam line flooding. Since some of the key operator actions differ substantially depending upon whether offsite power is available or not, the operators should be familiar with the correct measures that must be taken to circumvent such adverse situations.

Detailed criteria developed for these guidelines are presented in Table 4.2. These criteria should be considered as appropriate pending final resolution of the generic issue related to ISL (GI-105) and the unresolved safety issues on SGTR (USI A-3, A-4 and A-5).

4.3 Reduce High Core-Damage Frequency Sequences

The major contributors to the core-damage frequency (CDF) for a PWR with a large-volume containment are identified in Appendix A. They are accident sequences from loss of component cooling water, small LOCA (including interfacing systems LOCA), and transients including SBO and ATWS. Thus, if the [illegible] accident sequences (or a subset of them) can be reduced, [illegible] can be reduced.

4.3.1 Component Cooling Water (Guideline 3)

An event initiated by loss of component cooling water (CCW) may lead to RCP seal LOCA sequences because of loss of RCP seal cooling, under the condition that high-pressure injection (HPI) fails when these pumps are also cooled by CCW. In some plants the high-pressure injection systems are cooled directly by the essential service water (ESW). RCP seal LOCA may also result from loss of the ESW since the loss of the ESW, if it is not recovered in time, would lead to failure of the CCW system. Thus, the criteria should also apply to the ESW system.

The contribution of this initiator to CDF can be reduced by reducing the frequency of loss of the CCW and ESW systems. Because of pump seal LOCA the CDF can also be reduced by making the reactor coolant pump (RCP) seal cooling less susceptible to failure (e.g., using alternate seal injection systems[4] or steam-driven and self-cooled charging pumps[5]) or by improving the seal designs (e.g., using a pneumatic seal which prevents leakage[6]).

Detailed criteria developed for this guideline are presented in Table 4.3 and are considered appropriate pending the resolution of the generic issue on RCP seal failure (GI-23).

4.3.2 Operator Response for Recirculation (Guideline 4)

As presented in Section A.1, the small LOCA initiator has been found to be an important contributor to CDF in PWRs with large-volume containments. Major contributors to the small LOCA sequences are operator failures in the recirculation switchover operation and common cause failures of the containment sump suction valves and the high pressure pump suction valves. The recirculation switchover operation requires several operator actions to realign flow paths after the low refueling water storage tank (RWST) alarm is given:

i) changing positions of several valves downstream of the low-pressure injection pumps,
ii) switchover for containment spray system to recirculation, including valving in service water to the residual heat removal (RHR) heat exchangers,
iii) isolation of ECCS from the RWST suction, and
iv) valving in CCW to the RHR heat exchangers.

Although the sequences involving loss of containment spray were found not to be important contributors to risk for Zion, a main contributor to such sequences in other PRAs is often related to operator error similar to the recirculation switchover operation.

The operator failure probabilities may be reduced if the operators are trained in such a way that several of these actions during switchover are an integral operation for a single function.

Detailed criteria developed for this guideline are given in Table 4.4 based on these considerations and on eliminating common cause failures.

4.3.3 Station Blackout (Guideline 5)

In most PRAs for light-water-reactors (LWRs), station blackout (SBO) sequences have been major or prominant contributors to the CDF. The accident sequences in SBO are usually characterized by two categories of events: the failure to recover ac power before battery depletion (defeating the turbine-driven auxiliary feedwater system (AFWS) pump) and the RCP seal LOCA (because of failures of seal injection flow and RCP thermal barriers resulting from loss of the CCW and service water systems).

As part of the effort to resolve the unresolved safety issue (USI A-44), the NRC is proposing to amend its regulations "to provide further assurance that a station blackout (loss of both offsite power and onsite emergency ac power systems) will not adversely affect the public health and safety."[7] For accident sequences, developed by an individual plant examination (IPE), which involve the loss-of-offsite power and onsite emergency power, the proposed SBO rule should be examined for applicability. The criteria associated with Guideline 5 are intended to emphasize the need to search for plant specific features and potential common cause failures which could disable systems required to work during an SBO. For individual plants which are found to have a vulnerability to SBO, the criteria given in Table 4.5 highlight the importance of proper emergency procedures and operator training in recovering from an SBO event.

4.3.4 Reactor Coolant System Feed & Bleed Cooling (Guideline 6)

The success of reactor coolant system (RCS) feed & bleed cooling (high-pressure injection (HPI) and steam relief through the RCS power operated relief valves (PORVs)) has a significant impact on reducing the CDFs associated with sequences involving AFWS failures. Successful RCS feed & bleed cooling hinges on proper operator actions as well as on operability of the HPI systems, the PORVs, and the PORV block valves. The operator must ensure sufficient injection by charging pumps or HPI pumps and manually open the required number of PORVs. Appropriate containment heat removal systems must also be available. The use of RCS feed & bleed as a backup to the AFWS is being addressed as part of the unresolved safety issue on decay heat removal (USI A-45).

Detailed criteria for this guideline are given in Table 4.6 and should be considered as appropriate pending resolution of USI A-45.

4.3.5 Low-Pressure Injection (Guideline 7)

Appendix A (Table A.6) indicates that failure of low-pressure injection in a large LOCA is a significant contributor to the CDF at Zion. The dominant cause for unavailability of the low-pressure injection system (LPIS) is a human failure to reopen the two serial motor-operated valves on the suction side of the LPIS pumps after testing, combined with failures to discover the wrong position of the valves.

Detailed criteria for this guideline are given in Table 4.7.

4.3.6 Reactor Coolant System Depressurization by Secondary Blowdown (Guideline 8)

One potentially effective means of making up core inventory in case the high-pressure injection systems (HPIS) are unavailable (either in injection or recirculation) is to depressurize the RCS using the steam generators (SGs). After the RCS has been depressurized via secondary heat removal, the LPIS can be actuated for the RCS makeup operation. This emergency procedure could have significant impact on reducing the CDFs for the small and medium LOCA sequences. To attain success in this procedure, the operator must open the SG atmospheric steam dump valves, maintain auxiliary feedwater (AFW) or main feedwater to the SG and have one of the LPIS trains available. Although the net risk reduction for these low-pressure injection procedures has not yet been assessed by the NRC, the potential looks promising and criteria have been developed pending further analyses.

Detailed criteria for this guideline are given in Table 4.8.

4.3.7 Anticipated Transients Without Scram (Guideline 9)

The guideline for anticipated transients without scram (ATWS) sequences assumes compliance with the NRC rule on "reduction of risk from ATWS events for light-water-cooled nuclear power plants."[9] Also, in the NRC rule on ATWS, "the Commission has concluded that a reduction in the frequency of challenges to plant safety systems should be a prime goal of each licensee, and the Commission believes that ATWS risk reduction can also be achieved by reducing the much larger frequency of transients which call for the reactor protection system to operate."[8] As indicated in Appendix A, ATWS is not a dominant contributor to CDF for most PWRs. One factor in these results is the credit given to manual scram and emergency boration using the charging system to deliver borated water to the reactor pressure vessel. Failures of manual scram and emergency boration are dominated by operator action failures. These operator actions are not specifically addressed in the ATWS rule.

Detailed criteria developed for this guideline are given in Table 4.9.

4.3.8 Support System Interdependencies (Guideline 10)

One of the primary benefits of performing a rigorous PRA is that the system interdependencies are modeled and are reflected in the results. However, not all PRA studies have performed in-depth interdependence analyses and therefore have not ferreted out all of the possible subtle interdependencies. This may have profound effects upon their results. A dependency is defined as the failure of one system leading directly or indirectly to the failure of another system.

An in-depth application of basic PRA methodology with respect to interdependencies yielded significant findings on a previously heavily studied PWR. To illustrate this point, reference is made to the BNL study of system interactions (support system interdependencies) at Indian Point Unit 3.[5] The major finding of that study was that a specific single station emergency battery could fail and among other things, negate the entire low-pressure injection function. The point to be emphasized here is that none of the numerous other studies and reviews of the Indian Point 3 design were able to detect

this important single failure nor did the BNL study until all the support systems were explicitly modeled, linked together (the fault tree linking approach[10]) and solved using the SETS computer code.[11]

NUREG-1150[12] has provided a thorough application of the latest PRA methods to five reference plants and the results point out numerous insights into the importance of specific design differences among the studied plants. However, the NUREG-1150 authors emphasize the importance of support system differences and the difficulty of extrapolating the result of one plant to another.

It is not sufficient to make a single overall dependence table of the front-line and support systems for a given plant and simply compare that to the reference plant. No two plants will have the same set of system interdependencies. It is known that in U.S. nuclear plants support systems vary widely from plant to plant even though the plants may be of a similar class and have the same set of front-line systems.

It is recognized that following the interdependency evaluation steps as outlined in Table 4.10, in a rigorous fashion, is a major undertaking. This fact, however, does not diminish its importance.

Based upon the dominance of the loss of component cooling water (CCW) sequences and the SBO sequences for most PWRs, it is recommended that detailed interdependency tables be constructed for loss of CCW and SBO sequences, with all dependencies conditioned upon the loss of CCW and upon the existence of SBO for various lengths of time, respectively. These tables should also explicitly identify all of the expected failure mechanisms (e.g., identify whether battery failure is because of loss of room cooling or charge depletion).

4.3.9 Flooding of Emergency Equipment (Guideline 11)

Although medium or small leakages can be adequately mitigated by the existing sumps or pumpback systems, large water leakages are of primary concern in flooding. Potential water sources for excessive water release into the turbine building or auxiliary building include the condensate storage tank, the reactor coolant system, the service water system, the refueling water storage tank, and the fire protection system storage tank. Some of the major equipment located in the auxiliary building may include emergency core cooling (ECC) pumps and their electrical control panels.

Flooding can be initiated by (1) a major maintenance and (2) breaks in the pressurized or non-pressurized part of piping or components. In item 1, "major maintenance" refers to those actions which would require dismantling of system components thus eliminating a barrier between large sources of water and the auxiliary building. Flooding can partly be prevented and/or mitigated through proper training and procedures. For example, once the flooding occurs, the operator should be able to follow the instructions for responding to the alarm to identify the source of the flood and isolate it before the water level in the compartment reaches a critical level. The operator should also know about alternative devices or equipment which can be utilized to provide coolant injection to the RPV in case the emergency core cooling (ECCS) systems equipment fail in the flood compartment.

Although this type of vulnerability to flooding was identified for only one PWR, it is believed that the concerns are of general applicability to other designs. Thus, it is recommended that all PWR plants be screened for flooding vulnerabilities

Detailed criteria developed for this guideline are given in Table 4.11.

4.4 Using the Guidelines and Criteria

Numerous investigations, including PRAs, have been performed for the reference plants and for similar plants by both the NRC and the nuclear power industry. The insights gained from many of the studies have been used in developing the guidelines and criteria contained in this report (including the other volumes relating to other plant types). The guidelines and criteria are issued to guide the analyst performing an IPE. This guidance is in the form of plant features, operator actions and the criteria for assessing those features and actions found to be helpful in reducing the overall risk for Zion and other PWR plants. Thus, the guidance is given to provide a resource in examining the subject plant to determine if the same, or similar, guidelines will be of value in reducing overall plant risk. These guidelines and criteria are intended to be used solely as guidance, but they may include (as a subset) some requirements generated by the NRC on generic issues.

4.5 References for Section 4

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2. SECY-86-76, "Implementation Plan for the Severe Accident Policy Statement and the Regulatory Use of New Source-Term Information," NRC/EDO, February 28, 1986.

3. "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," U.S. Nuclear Regulatory Commission, NUREG-0844, Draft Report for Comment, April 1985.

4. E. Silvestri et al., "A Model for the Probability of Core Uncovery in LOOSP Induced Accidents, as Applied in the Probabilistic Safety Study for ENEL PWR Standard Power Plant," Paper No. 109, Proceedings of the International ANS/ENS Topical Meeting on Probabilistic Safety Methods and Applications, Volume 2, February 1985, San Francisco, CA.

5. G. Edison, "Sizewell-B: Analysis of British Application of U.S. PWR Technology," NUREG-0999, May 1983.

6. H. L. Schnurer and H. G. Seipel, "The Safety Concept of Nuclear Power Plants in the Federal Republic of Germany," Nuclear Safety, 24, 743 (1983).

7. NRC Station Blackout Proposed Rule, Federal Register, Volume 51, No. 55/March 21, 1986, pgs. 9829-9835.

8. ATWS Final Rule - Code of Federal Regulations, Title 10, Section 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram Events for Light-Water Cooled Nuclear Power Plants," June 1984.

9. R. Youngblood et al., "Fault Tree Application to the Study of Systems Interactions at Indian Point 3," Brookhaven National Laboratory, NUREG/CR-4207, January 1986.

10. American Nuclear Society and Institute of Electrical and Electronics Engineers, "A PRA Procedures Guides," NUREG/CR-2300, January 1983.

11. R. B. Worrell and D. W. Stack, "A SETS User's Manual for the Fault Tree Analyst," Sandia National Laboratories, NUREG/CR-0465, SAND77-2051, November 1978.

12. "Reactor Risk Reference Document," U.S. Nuclear Regulatory Commission, NUREG-1150 - Draft for Comment, February 1987.

Table 4.1 Criteria for PWR Large-Volume Containments
Guideline 1: Maintain Containment Integrity

Concern: Breach of the containment boundary in the progression of a severe accident can lead to significant releases of radioactivity.

Functions: Containment Spray (Guideline 1.A)
- Containment Heat Removal
- Fission-Product Scrubbing
- Debris Bed Cooling

Filtered Venting (Guideline 1.B)
Assessment of Early Containment Failure (Guideline 1.C)

Guideline 1.A. Provide Containment Spray and Long-Term Heat Removal

Basis: Early initiation of containment spray will help flood the reactor cavity, aid in fission-product decontamination and cool the core debris as it exits from the reactor vessel. Continued operation of the containment spray or fan coolers will remove heat from the containment atmosphere and delay containment failure.

Caution: Failure of recirculation has been identified as a dominant contributor to core damage for some PWRs. For small break LOCA sequences, early initiation of core spray may deplete the RWST and limit the available RCS injection capability if recirculation is unsuccessfully (See Guideline 4.A.).

Criteria:

The following should be assessed to ensure containment heat removal capability:

1.A.1. The heat removal provided by the containment spray related components should be sufficient to remove heat loads anticipated during the dominant accident sequences. These loads include but are not limited to decay heat and the chemical energy released from metallic oxidation.

1.A.2. Containment spray should commence when the containment pressure exceeds a predetermined value calculated in accordance with the appropriate PWR Emergency Response Guidelines (ERGs).

1.A.3. Containment spray should be terminated when the containment pressure decreases below a predetermined value calculated in accordance with the ERG.

Guidance: Alternate sources of containment spray such as a diesel-driven fire pump should be assessed for their capability to provide sufficient flow and head for adequate containment heat removal.

1.A.4. The ability of equipment designated for containment spray to function in a reliable manner under the predicted containment conditions associated with sequences for which operation of the containment spray is needed should be assessed.

1.A.5. Operator training and emergency procedures should specify the flow paths and specific components to be aligned and their required positions for initiating containment spray. If a backup system and/or equipment are to be utilized, operator training and procedures should identify the flow paths and specific actions required for any temporary system cross connections.

1.A.6. Operator training and emergency procedures should specify the plant parameters that will prompt the operators to initiate and terminate the containment spray. Training and procedures should be consistent with the time required to align the system and components as required.

The following should be assessed in order to evaluate the capability of the containment spray to reduce fission-product contamination of the containment atmosphere:

1.A.7. The spray system should be assessed for its ability to cover the entire containment volume for an adequate time, with spray droplets of an appropriate size. Such an assessment would include the total amount of water available for long-term spray operation, as well as the pressure under which the water could be supplied to the spray headers by various sources. The elevation of the spray, the nozzle spray pattern, as well as large obstructions below the spray headers, should be considered when assessing volume coverage.

In addition, to promote debris bed cooling the following criteria should be assessed to ensure flooding of the reactor cavity before reactor pressure vessel (RPV) failure:

1.A.8. The spray initiation point (Criterion 1.A.2) should be assessed to ensure that the sprays will be initiated early enough to flood the reactor cavity before RPV failure for the dominant accident sequences.

1.A.9. The spray termination point (Criterion 1.A.3) should be assessed to ensure that spray termination will not allow the debris bed to reheat after it has been successfully cooled.

Guideline 1.B. Provide Filtered Venting

Basis: For those plants where ac independent containment sprays and long term containment heat removal are not available a filtered vent path should be considered to ensure fission-product mitigation for late containment failure events.

Caution: Containment venting should not be indiscriminantly performed. A clear understanding of the accident sequence in progress should have been attained prior to initiating venting. The effects of venting should have been assessed and made known to the operators during the training program. The assessment should include the effects of containment venting on the operation of emergency core cooling (ECC) injection systems and health consequences.

Criteria:

The following should be assessed to ensure filtered venting capability:

1.B.1. For accident sequences where filtered venting has been assessed to be beneficial, filtered venting should commence when containment pressure reaches the predetermined containment venting pressure set point. In selecting the containment venting pressure set point, the following functions should be ensured:

a. The ultimate containment pressure capability would not be exceeded.

b. The backpressure acting on the safety relief valve assemblies would not prevent them from performing their function.

c. The vent valve assemblies would not be prevented from performing their function.

1.B.2. If local initiation of filtered venting is necessary, the time required to perform this function and the potential for exposing personnel to the harsh environment should be taken into account in the training and procedures.

1.B.3. Operator training and emergency procedures should specify the plant parameters that will prompt the operators to make preparation, commence and terminate the venting sequence. The training and procedures should also be consistent with the required actions and timing of those actions so venting will commence immediately when required (see Criteria 1.B.1 and 1.B.2). The training and procedures should further specify the flow path(s) available for venting, specific components to be aligned, and the required positions/states for these components. The training and procedures should specify how to proceed if termination of venting is not possible.

1.B.4. For each accident sequence where venting is credited (e.g., assumed to prevent containment failure) the capacity of the vent lines and associated vent valves should be assessed to determine whether the venting capacity has the capability to decrease containment pressure.

1.B.5. The filtering media should be capable of reducing the released fraction of the radioactivity of the non-noble gas component by an order of magnitude.

1.B.6. The capability of equipment used to support venting to function under predicted environmental and fluid loads associated with severe accident sequences (including station blackout) should be assessed to determine whether the equipment would be available for an appropriate time of operation including the vaporization release phase of core-concrete interaction.

Table 4.1 (Continued)

1.B.7. The filters should be capable of accepting the aerosol loading from core-concrete interactions while remaining functional.

Guideline 1.C. Assessment of Early Containment Failure

Basis: The SARRP uncertainty analyses for Zion and Surry indicate that the likelihood of early containment failure for core melt accidents is about 5% (mean value). These early failures are because of the combined effects of direct heating and hydrogen detonation or deflagration. Even at this low likelihood the predicted releases from early containment failure have been assessed to be a dominant part of the risk for both plants.

Criteria:

1.C.1. The applicability of the containment performance for Zion and Surry to the individual plants should be assessed. Specific structual analysis for similar containment design (e.g., for steel shell large-volume containments) may be necessary to determine the ultimate pressure capability of each containment.

1.C.2. The likelihood of early containment failure because of direct heating for the dominant core-damage sequences should be determined.

Guidance: If the likelihood of early containment failure is substantially larger than at Zion then additional mitigation should be considered (e.g., containment sprays will aid in fission-product decontamination and flooding for some reactor cavity geometries may reduce direct heating).

1.C.3. The likelihood of early containment failure because of hydrogen detonation or deflagration for the dominant core-damage sequences (GI-124) should be determined.

Table 4.2 Criteria for PWR Large-Volume Containments
Guideline 2: Interfacing Systems LOCA
(Including Steam Generator Tube Rupture)

Concern: The interfacing systems LOCA sequences represent potentially high release sequences and were found to contribute significantly to risk, even though their contribution to core-damage frequency (CDF) is not significant. The contribution to CDF by the steam generator tube rupture (SGTR) sequences is also relatively small; however, the associated risk can be potentially significant because the leakage of primary reactor coolant to the secondary side of the steam generator (SG) will cause release of radioactivity to the outside of containment via the steamlines, the turbines, the condenser and the SG atmospheric steam dump and safety valves.

Function: Maintain Primary System Integrity (Guidelines 2.A and 2.B)

Guideline 2.A. Prevent Overpressurization of Low Pressure Systems

Basis: Implementation of the following criteria will ensure the frequency of an interfacing systems LOCA will remain acceptably low.

Note: Resolution of the generic issue (GI-105), which deals with interfacing systems LOCAs for both BWRs and PWRs, may impact this guideline. Therefore, the criteria below should be considered as appropriate pending resolution of the GI-105.

Criteria:

2.A.1. All low-pressure lines that potentially could be overpressurized should be identified and should be provided with alarms to alert the operator of an overpressure event.

2.A.2. The equipment designated to provide isolation and prevent overpressurization, such as the residual heat removal (RHR) line isolation valves or the low pressure injection system check valves, should periodically undergo operability testing and local leak rate testing (LLRT).

2.A.3. The relief capability of the relief valves designated to mitigate low-pressure system overpressurization should be established. In most, if not all, cases these relief valves were not sized with the possibility of an interfacing systems LOCA in mind. However, given that an interfacing systems LOCA occurs in a non-isolatable portion of a low-pressure system, there may be alternatives available to the operator such as taking advantage of additional relief valves. If such or similar actions are found to be helpful, they should be factored into the appropriate emergency procedures.

2.A.4. Operator training and procedures should specify the actions to be taken to isolate the low-pressure systems identified above or depressurize the primary system, thereby mitigating the consequences of the interfacing systems LOCA.

2.A.5. After each reactor shutdown and cooldown, testing of the isolation function of the pressure isolation valves should be performed. Testing of these valves should not be performed under reactor operating conditions.

Guideline 2.B. Prevent Steam Generator Tube Rupture (SGTR) and Minimize Its Consequences

Basis: The following criteria are provided to ensure that the risk because of SGTR events at each plant is a small fraction of the overall risk (See NUREG-0844). These criteria should be considered as appropriate pending resolution of the unresolved safety issues related to SGTR (USI A-3, A-4 and A-5).

2.B.1. The structural integrity of steam generator (SG) tubes should be maintained by protecting them against all plausible causes leading to weakening, cracking or bursting of the tubes. As part of the resolution of the unresolved safety issues related to SGTR (USI A-3, A-4 and A-5) the NRC staff has issued Generic Letter 85-02 to ensure that each licensee's SG tube integrity program complies with staff recommendations.

2.B.2. Operator training and procedures should specify the means by which the occurrence of SGTR can be correctly diagnosed, the faulty SG identified and isolated in a timely manner, and the necessary recovery actions taken. These actions include reactor coolant system (RCS) cooldown to below the saturation temperature corresponding to the faulty SG pressure by dumping steam only from the intact SGs to establish sufficient subcooling margin, subsequent RCS depressurization to the faulty SG pressure with establishment of sufficient RCS inventory and safety injection termination to avoid repressurization and further primary-to-secondary leakage.

2.B.3. If offsite power is unavailable during the recovery actions, the RCS cooling described under 2.B.2 above should be achieved by using the atmospheric relief valves on the intact SGs, since neither the turbine bypass valves nor the main condenser would be available. Also, RCS pressure should be controlled by using pressurizer PORVs or auxiliary spray since normal pressurizer spray will be unavailable because of loss of the reactor coolant pumps.

Table 4.3 Criteria for PWR Large-Volume Containments
Guideline 3: Component Cooling Water

Concern: An event initiated by loss of component cooling water (CCW) or essential service water (ESW) may lead to Reactor Coolant Pump (RCP) seal LOCA sequences because of loss of RCP seal cooling and seal injection, under the condition that high pressure injection also fails. This is an important contributor to CDF at Zion, as is also the case at other PWRs. RCP seal failure and CCW cooling loss are currently designated as generic issues by the NRC (GI-23 and GI-65, respectively).

Function: Adequate Cooling of Engineered Safety Features Systems (Guideline 3.A)

Guideline 3.A. Provide Adequate Cooling to Engineered Safety Features Systems

Basis: Implementation of the following criteria on CCW will significantly reduce the CDF and risk because of RCP seal LOCA sequences. Loss of CCW initiated by loss of electric power is addressed by Guideline 5: Station Blackout. If, depending on the design configuration of a particular plant, loss of ESW is likely to result in an RCP seal LOCA, then the criteria should be applied to the ESW system.

Criteria:

3.A.1. Operator training and procedures should specify actions to be taken to shed nonessential loads requiring CCW (or ESW) to increase the length of time that the plant can cope with partial loss of CCW (or ESW).

3.A.2. Special emphasis should be placed upon the review of the CCW (or ESW) system to assure that common cause failures have been eliminated to the extent practical from the design.

3.A.3. The capability of the CCW (or ESW) system and associated equipment to function under predicted environmental and fluid loads associated with severe accident sequences should be assessed to determine whether the equipment would be available for an appropriate time of operation.

3.A.4. Alternate injection systems to cool the RCP seals or improved seal designs which prevent leakage should be considered for those plants where RCP seal LOCAs comprise a dominant part of the CDF.

Table 4.4 Criteria for PWR Large-Volume Containments
Guideline 4: Operator Response for Recirculation

Concern: Inadequate decay heat removal because of failure in the recirculation switchover operation has been found to be a leading contributor to severe accident sequences, in particular, sequences initiated by small LOCAs. Failure in the recirculation switchover of the high pressure systems leads to core damage because of insufficient coolant makeup and thus inadequate decay heat removal from the core. Failure in the recirculation switchover of the containment spray system and failure to establish a containment spray path from RHR lead to inadequate containment heat removal (CHR). Containment failure because of inadequate CHR could create net positive suction head (NPSH) problems for the pumps taking suction from the containment sump, which could in turn lead to recirculation failure and subsequent core damage.

Functions: Adequate Core Recirculation Cooling (Guideline 4.A)
Adequate Containment Heat Removal (Guideline 4.B)

Guideline 4.A. Provide Adequate Core Recirculation Cooling

Basis: Implementation of the following criteria will significantly reduce the CDF and risk because of core damage associated with the recirculation cooling phase.

Criteria:

4.A.1. Operator training and procedures should specify the actions to be performed to realign flow paths, including recovery actions that must be performed during recirculation switchover (especially at high RCS pressure) and RHR heat exchanger injection cooling. These actions should be specified in an integrated fashion, so that no single operator has a designated role whose incorrect action would cause loss of the switchover or cooling functions.

4.A.2. Training and procedures should attempt to avoid failure of both injection pumps because of switchover to a sump which does not provide sufficient NPSH to operate the pumps. Consideration should be given to providing procedures which will specify refill of the RWST to provide continued primary injection capability if recirculation switchover fails.

Guideline 4.B. Provide Adequate Containment Heat Removal (CHR)

Basis: Implementation of the following criteria will significantly reduce the CDF and risk associated with containment failure.

Criteria:

4.B.1. Operator training and procedures should specify all actions (including alternatives) required to initiate and maintain CHR under severe accident conditions when recirculating from the containment sump.

Criteria:

4.B.1. Operator training and procedures should specify all actions (including alternatives) required to initiate and maintain CHR under severe accident conditions when recirculating from the containment sump.

4.B.2. When manual-remote operation of equipment is required to initiate CHR, the operator training and procedures should specify the actions for performing these functions and provide an understanding of and experience with the time required to perform them.

4.B.3. The capability of equipment designated for CHR to function under predicted environmental and fluid loads associated with severe-accident sequences should be assessed to determine whether the equipment would be available for an appropriate time of operation. Note that this criterion does not imply that the equipment will be qualified to a new design envelop.

Table 4.5 Criteria for PWR Large-Volume Containments
Guideline 5: Station Blackout

Concern: Station blackout sequences have been shown to be one of the leading classes of severe-accident sequences in terms of both CDF and risk.

Function: Reliable Operator Response (Guideline 5.A)
Post Accident Decay Heat Removal Capability (Guideline 6.A)

Guideline 5.A. Provide Operator Response During Station Blackout

Basis: Significant study and research have preceded current work on severe accidents; in particular, reference is made to the rulemaking activity already under way on station blackout. It is assumed that when the station blackout rule is finalized, some requirements of the rule may be similar in form to the criteria below. Nevertheless, during an individual plant examination (IPE), it is important to highlight those areas which previous PRAs found to be important contributors to the station blackout core-damage frequency. In addition, auxiliary feedwater system (AFWS) turbine-driven pump reliability is being addressed in GI-124 and the following related criteria should be considered appropriate pending resolution of this generic issue.

Criteria:

5.A.1. Operator training and procedures should specify the actions that should be performed to provide core cooling and decay heat removal and should specify the systems and components to be aligned and their required positions.

5.A.2. Operator training and procedures should specify the actions required to restore offsite and onsite emergency ac power prior to depletion of the dc power supply during station blackout.

5.A.3. Operator training and procedures should specify the actions that should be performed to shed nonessential loads requiring dc power to increase the length of time that the plant can cope with a station blackout.

5.A.4. Operator training and procedures should specify any actions that should be taken prior to the loss of dc power given that ac power has not been restored.

5.A.5. Operator training and procedures should specify actions required to initiate operation and/or assure that containment spray is operable under station blackout conditions. If containment spray is lost under possible degraded core conditions, specific procedures should address the conditions under which containment spray can be restored (under some conditions, restoration of sprays could deinert the containment and result in a hydrogen burn).

Table 4.5 (Continued)

5.A.6.	The capability of a turbine-driven auxiliary feedwater train or its equivalent to function under predicted environmental and fluid loads associated with severe accident sequences (including station blackout sequence) should be assessed to determine whether the equipment will be available for an appropriate time of operation.
5.A.7.	Special emphasis should be placed upon the review of the AFWS to assure that common cause failures have been eliminated to the extent practical from the design.

Table 4.6 Criteria for PWR Large-Volume Containments
Guideline 6: Reactor Coolant System (RCS)
Feed & Bleed Cooling

Concern: For many PWRs, RCS feed & bleed cooling can be employed to remove the decay heat in case the RCS heat removal via the SGs is insufficient. If the feed & bleed cooling also is inadequate, core uncovery would ensue.

Functions: Provide Alternate Core Heat Removal Capability (Guideline 6.A)

Guideline 6.A. Provide Operator Response for RCS Feed & Bleed Cooling

Basis: Fulfilling the following criteria will improve the chances for successful RCS heat removal subsequent to loss of the main and auxiliary feedwater systems. Overall decay heat removal capability is being addressed by USI A-45.

Criteria:

6.A.1. Assess the capacity of the power operated relief valves (PORVs) or head vent valves and the high-pressure injection system to remove decay heat.

6.A.2. Operator training and emergency operating procedures should specify the actions required for RCS feed & bleed cooling. In particular, the diagnostic means of determining the need for feed & bleed cooling and the detailed procedures to be followed as well as the number of PORVs to be manually opened should be clearly stipulated. Appropriate means of containment heat removal should also be available (Guideline 4.B).

Table 4.7 Criteria for PWR Large-Volume Containments
Guideline 7: Low-Pressure Injection

Concern:	Failure of low-pressure injection in a large LOCA was found to be a significant contributor to core-damage frequency for Zion.
Functions:	Low Pressure Injection (Guideline 7.A.)

Guideline 7.A. Improve Availability of Low-Pressure Injection

Basis: Implementation of the following criteria will enhance the core inventory makeup availability of the low-pressure emergency coolant injection function when called upon.

Criteria:

7.A.1. Surveillance procedures for the low-pressure injection system (LPIS) should ensure that potentially mispositioned valves in the suction side of LPIS pumps will be identified and restored to normal as part of restoration after testing.

7.A.2. Operator training and procedures should ensure operator recognition of mispositioned LPIS valves and should specify the actions to be performed to establish the required flow paths in the suction of the LPIS pumps.

Table 4.8 Criteria for PWR Large-Volume Containments Guideline 8: Reactor Coolant System (RCS) Depressurization by Secondary Blowdown Blowdown

Concern: Small or medium LOCAs followed by failure of the high pressure recirculation or the high pressure injection system were determined to be important contributors to CDF for Zion. Some of these transients were also found to contribute significantly to risk.

Function: Depressurize the Reactor Coolant System (RCS) (Guideline 8.A)

Guideline 8.A. Provide RCS Depressurization Capability

Basis: Implementation of the following criteria will enhance the core inventory makeup capabilities of the low-pressure emergency coolant injection and recirculation function in the event that the high pressure systems are unavailable.

Criteria:

8.A.1. Operator training and procedures should specify the actions to be performed to realign flow paths, initiate and control RCS depressurization by secondary blowdown in order to provide emergency core makeup via the low-pressure injection system (LPIS) if the high-pressure injection systems (HPIS) are unavailable. The procedures, such as opening the SG atmospheric relief valves and maintaining auxiliary feedwater/emergency feedwater or main feedwater to the SGs, should follow the instructions given in the proper emergency response guidelines and account for the possibility of steam generator tube rupture (SGTR - Guideline 2.B).

8.A.2. The capability of equipment and systems designated to depressurize the RCS by secondary blowdown to function under predicted environmental and fluid loads associated with severe accident sequences should be assessed (this does not imply that the equipment will be qualified to severe accident conditions) to determine whether the equipment would be available for an appropriate time of operation.

Table 4.9 Criteria for PWR Large-Volume Containments
Guideline 9: Anticipated Transients
Without Scram (ATWS)

Concern: Although PRAs for some PWRs indicate that ATWS sequences are not significant contributors to core-damage frequency, they have been studied extensively in the past because they may lead to plant conditions that may render several engineered safety features ineffective and thus may contribute significantly to risk.

Function: Reliability of the Reactor Protection System and Operator Response During ATWS (Guideline 9.A.)

Guideline 9.A. Provide Operator Response During ATWS

Basis: The criteria developed here are based on the assumption that each of the plants is (or will be) in compliance with the ATWS final rule dated July 26, 1984. PRA studies have shown that the predicted core-damage frequency because of ATWS is significantly lowered based upon modifications which comply with the ATWS rule. The major thrust of the ATWS rule is on the addition and/or upgrading of scram related systems and equipment to prevent an ATWS. Human reliability studies performed in support of NUREG-1150 point to potential benefits for improved operator training and procedures to mitigate the effects of an ATWS and prevent core damage from occurring. For any individual plant which may be found to be vulnerable to ATWS, the following criteria reflect added measures that emphasize the operator's role and function in mitigating an ATWS initiator.

In the case when the automatic scram system fails during plant transients, the operator is required to attempt to manually scram the reactor and inject borated water to the RPV by using the charging system including repositioning the valves which are necessary to properly align the charging system to the boron supply.

Criteria:

9.A.1. Operator training and procedures should specify the actions required for the manual scram and emergency boration. They should also specify the responsibilities of operating staff crew members and clarify how information will be exchanged among them. In particular, instrumentation readings may have to be relayed between the crew members(s) operating the control boards and the senior reactor operator coordinating the crew's response to accident.

9.A.2. The capability of systems and equipment designated for operation during an ATWS to function under predicted environmental and fluid loads associated with severe accident sequences should be assessed to determine whether the equipment would be available for an appropriate time of operation.

Table 4.10 Criteria for PWR Large-Volume Containments
Guideline 10: Support System Interdependencies

Concern: When conducting a PRA, IPE or similar analysis, it is imperative that the support system interdependencies be fully developed, understood and reflected in the final results. Otherwise there is no assurance that the dominant core-damage/risk sequences have been identified.

Function: Support System Interdependencies (Guideline 10.A)

Guideline 10.A. Examine Support System Interdependencies

Basis: Implementation of the following criteria will ensure that the full set of support system interdependencies have been identified and have been reflected in the results.

Note: The following criteria are easily outlined but are not easily implemented. The complex nature of a nuclear power plant makes it imperative that this area of analysis be fully examined. However, since no two plants have identical support systems, this analysis should be done on a plant specific basis.

Criteria:

10.A.1 All systems that provide any direct support to either a frontline or support system should be identified along with its supported system. For each dependency that is identified, the failure mechanism and time should be estimated.

10.A.2. Each dependency should be conditioned as appropriate as to what sequences or under what (if not all) circumstances it applies. In view of its importance, a separate station blackout dependency table should be provided which gives the available systems, their anticipated survival period and the ultimate cause (e.g., no room cooling) of their failure.

10.A.3. As in Criterion 10.A.2, a dependency table should be provided which shows the effects of loss of component cooling water (or service water) on other systems.

10.A.4. The dependencies should then be linked together (preferably by computer) within the analysis in order that the extent to which their influence reaches through the systems to a consequence will be discovered. This will help identify secondary dependencies to ensure that no one failure in a support system has any unknown critical outcome on other support or front line systems.

Table 4.11 Criteria for PWR Large-Volume Containment Guideline 11: Flooding of Emergency Equipment

Concern: An excessive water release into a portion of the turbine building or auxiliary building outside the primary containment which houses a concentration of safety-related equipment raises the possibility of a common-mode event disabling key equipment. At least one plant with a large-volume containment has been identified in which the location of safety equipment makes the internal flooding initiator a substantial contributor to CDF.

Function: Prevent or Mitigate Internal Flooding (Guideline 9.A)

Guideline 11.A. Prevent or Mitigate Internal Flooding

Basis: Implementation of the following criteria may reduce the potential of a common-mode failure of safety equipment because of internal flooding for PWR plants where the systems' layout combines important safety equipment in compartments with exposure to possible inundation.

Criteria:

11.A.1. For those areas identified as vulnerable to flooding, instrumentation should be added to alert the operator of such an occurrence.

11.A.2. Operator training and procedures should ensure that the operator will respond to isolate any internal floods that occur.

11.A.3. Operator training and procedures should ensure that the operator is prepared to use alternate injection sources still available if flooding causes a common-mode failure of ECCS equipment.

11.A.4. The electrical systems should be assessed for the possibility of cascading failures because of flood-induced electrical shorts. Additional isolation devices (e.g., circuit breakers) should be considered, if needed.

11.A.5. The addition of water-tight doors/walls might be effective ways to lessen the common-mode threat. Water-tight doors between compartments should be alarmed to notify the operator when they have been left open.

APPENDIX A

SEVERE ACCIDENT RISK INSIGHTS

This appendix provides a review of the Industry Degraded Rulemaking Program (IDCOR) and the U.S. Nuclear Regulatory Commission (NRC) Severe Accident Research Program (SARP) analyses for the Zion plant which is a pressurized water reactor (PWR) with a large-volume containment. Differences between these studies and related studies are identified and insights are provided which helped in the development of the plant type-specific guidelines and criteria for preventing and mitigating severe accidents which are discussed in Section 4 of this report. In addition to the review of the IDCOR and SARP analyses for Zion, design features of other plants (Surry, Oconee) with a similar containment are also briefly surveyed to identify similarities and differences among the plants. The insights gained from this overview of the design features also contributed to the development of the guidelines and associated criteria provided in Section 4.

The Zion Station Units 1 and 2, located in Lake County, Illinois, are Westinghouse PWRs of thermal output of 3250 MW per unit. Table A.1 summarizes the design of the safety-related systems at Zion in comparison with those at Surry and Oconee. Figure A.1 shows a heat and fluid flow diagram for the Zion's various cooling systems.

Section A.1 describes the review of the various estimates of the core-damage profile. Core-meltdown phenomena and containment response are addressed in Section A.2. Differences between IDCOR and SARP estimates of fission-product release and offsite consequences are discussed in Sections A.3 and A.4. Finally, Section A.5 indicates the insights obtained from the review of these studies.

A.1 Core-Damage Profile

The main objective of this section is to present within the scope of this study, the Zion core-damage profiles emerging from the IDCOR[1] and the NRC Accident Sequence Evaluation Program (ASEP)[2] analyses.

The primary information reviewed in this section is from Refs. 1 and 2. The information in Refs. 3 through 5 was also used in the review. The scope of the SARP analysis for Zion[2] is a limited rebaselining of the Zion review.[3] The rebaselining effort on Zion is limited to the models and accident sequences addressed in the Zion review. Thus, the results in the SARP analysis represent an update of the existing dominant sequences in the Zion review, and no attempt was made to identify any new dominant sequences which may result from plant changes or improved PRA methods.

A.1.1 IDCOR Baseline Estimate of Zion Core-Damage Frequency

This section presents a brief summary of the major differences between the dominant core-damage sequences designated by the Zion IDCOR baseline risk profile (or the Zion pre-IDCOR risk profile) and those identified in the course of reviewing and evaluating the Zion Probabilistic Safety Study (ZPSS) by Sandia National Laboratories (SNL)(NUREG/CR-3300, hereafter referred to as the Zion review). The risk profile being considered here consists of two

distinctive parts, one displaying the plant core-damage frequency (CDF) and the other the societal risk and population dose. Both the Zion IDCOR baseline risk profile and the Zion pre-IDCOR risk profile derive their origins from the ZPSS. They differ chiefly in the part pertaining to plant risk and release consequences. As far as the dominant core-damage sequences and their frequencies are concerned, no distinction can be made between these two Zion IDCOR risk profiles. This is mainly because of the fact that when the Zion IDCOR baseline risk profile was developed by partially updating the ZPSS, the revision was carried out primarily to reflect new information on accident processes and containment responses. No alterations were made to initiating event frequencies, success criteria, system unavailabilities, or plant configuration. The dominant core-damage sequences and their frequencies appearing in both the Zion pre-IDCOR risk profile and the Zion IDCOR baseline risk profile are, therefore, completely identical. Incorporation of the new information in accident process analysis and containment response, however, results in a significant decrease of societal latent cancer fatality risk (by a factor of about 60 as compared to the Zion pre-IDCOR risk profile) and no early fatalities for the Zion IDCOR baseline risk profile.

For reference, the Zion IDCOR baseline dominant accident sequences and the associated CDFs are listed in Table A.2. This list is similar to that presented in the ZPSS (ZPSS Table 8.10-1), except that the two anticipated transients without scram (ATWS) sequences, originally numbered 5 (loss of main feedwater ATWS) and 6 (turbine trip ATWS), are replaced by two new sequences, denoted 5-TEFC and 6-TEC, respectively, both of which are initiated by fires. This was done, in part, to conform to the changes made in Revision 1 to the ZPSS and also to reflect the potential for the two new sequences to contribute significantly to CDF. Besides these difference, two new sequences, 17-AE and 18-SLF, are added to the Zion IDCOR baseline list. Also, a slight discrepancy in the result of CDF quantification can be found for some of the accident sequences, as footnoted in Table A.2.

It can be observed that, for the Zion IDCOR baseline risk profile, the most dominant core-damage sequence is the small loss-of-coolant accident (LOCA) with recirculation failure, 1-SLFC (approximately 39% of the overall CDF). This is followed by the seismic sequence, 2-SE (about 13%), and large and intermediate LOCAs with recirculation failures, sequence 3-ALFC and 4-ALFC, each contributing about 12%.

A.1.2 Zion Review Estimate of Zion Core-Damage Frequency

In contrast with the Zion IDCOR baseline risk profile discussed above, the dominant accident sequences generated in the Zion review deviate substantially from those presented in the ZPSS (ZPSS Table 8.10-1). The chief objective of the Zion review was to scrutinize significant omissions and crucial judgements which might have been made in the ZPSS, and to assess their impact on the quantitative results of the ZPSS. This was achieved basically by reviewing the initiating events, the event trees depicting the plant system response to the initiating events, the system fault trees, the human reliability analysis, and other probabilistic risk assessment (PRA) methodology employed in the ZPSS. A total of 14 dominant accident sequences were identified in the Zion review; they are tabulated in Table A.3 for quick reference. As compared with the list of dominant accident sequences presented in the ZPSS (ZPSS Table 8.10-1) or that displayed in Table A.2 (for the Zion IDCOR

baseline), the most remarkable difference is the identification of several dominant accident sequences involving failure of the component cooling water (CCW) system. In fact, the most dominant accident sequence was determined to be CCW failure leading to failure of all charging and safety injection (SI) pumps and the development of reactor coolant pump (RCP) seal LOCA. The next three dominant positions are taken up consecutively by three accident sequences initiated by loss of offsite power, followed by loss of component cooling water, with failure to restore offsite power in four hours, one hour, and eight hours (with simultaneous failure of containment fans), respectively. The sequence of small LOCA with recirculation cooling failure, ranked first in the Zion IDCOR baseline core-damage ranking list, is dropped to the fifth position on the Zion review list. Only 6 (including the interfacing systems LOCA sequence) out of the 18 accident sequences, deemed dominant in the Zion IDCOR baseline core damage list, appear in the Zion review dominant core-damage list.

In addition to the CCW system failure sequences, other important accident sequences newly identified in the Zion review include failure of dc Bus 112 with loss of auxiliary feedwater (AFW), and loss-of-offsite power accompanied by failure of AFW and the loss of "feed and bleed" capability. One major reason for the sudden emergence of CCW system failure as predominant accident sequences in the Zion review is the recognition of the possibility for failure of both charging and SI pumps shortly after the loss of CCW. Such a possibility was not considered credible in the ZPSS analysis, which assumed that loss of CCW alone would not lead directly to core melt without additional system failure.

In the following, a condensed summary is given on the CDF quantifications performed by the Zion review for each of the dominant accident sequences identified. It should be remarked that, for accident sequences involving loss of CCW the system success criteria is defined to be 2 CCW pumps (out of 5 CCW pumps) and 2 service water system (SWS) pumps (out of 6 SWS pumps) operating.

(1) Failure of Component Cooling Water (CCW), SEFC

The CDF for this sequence is quantified based on the presumption that failures of both charging pumps and SI pumps will ensue shortly after the loss of CCW. The sequence is initiated by loss of CCW with consequential loss of component cooling to the RCP thermal barriers. Seal cooling to the two centrifugal charging pumps, normally placed to service in succession, will fail in about ten minutes after the onset of CCW failure. All four RCP seals fail in about 30 minutes following the loss of seal cooling, with coolant leakage rate of about 300 gallons per minute per pump. Both SI pumps are actuated by low reactor coolant system (RCS) pressure and fail in about five minutes because of lack of cooling. Core uncovery will occur as a result of the loss of makeup capability through either the charging or safety injection pumps. Core melt is predicted unless cooling to the SI pumps is restored in about 45 minutes. The frequency for complete loss of CCW is given in the ZPSS as 9.4(-4)* per reactor year. Of this, the Zion review estimated that 2.3(-4) is because of pipe rupture. By applying a recovery factor of 0.5, CDF because of CCW failure attributable to pipe rupture is estimated to be 1.2(-4). The frequency of CCW loss not caused by pipe rupture is 7.1(-4). For these

*9.4(-4) = 9.4×10^{-4}.

events, failure of recovery is defined as failure to start the two standby pumps, each with an unavailability of 0.033, which accounts for that because of maintenance and pump failing to start. CDF because of failure to recover from loss of CCW not originated from pipe rupture is, accordingly, 7.1(-4) (0.066) = 4.7(-5). Therefore, the overall CDF because of loss of CCW is about 2(-4). The basis for this CDF is controversial and the SARRP[11] study has revised the estimate to 5.5×10^{-6}.

(2) Loss-of-Offsite Power: Loss of Component Cooling Water: Failure to Recover Power in Four Hours, SEFC

The accident sequence envisaged here is initiated by loss-of-offsite power followed by loss of CCW with failure to restore offsite power in four hours. As described above in (1), loss of CCW can induce a series of events leading to a seal LOCA with loss of makeup capability and the eventual core damage.

The initiating event frequency for loss-of-offsite power is given in the ZPSS as 5.7(-2). To this are added the frequencies (taken from the ZPSS) of turbine trip, loss of main feedwater and reactor trip, each followed by loss of offsite power. The sum of these frequencies totals 0.061, which is taken to be the initiating frequency for loss of offsite power in the Zion review. Simultaneous loss-of-offsite power to both Zion Units 1 and 2 is assumed.

The probability for failure to restore offsite power in four hours (but, success prior to eight hours) was estimated to be 0.15. The conditional probability of CCW failure, given loss-of-offsite power, (CCW)', is dependent on the state of the vital ac buses. Calculation of these probabilities for each of the eight electric power states is detailed in the Zion review. The conditional probabilities of the electric power states following a loss of offsite power initiating event, (EP)', are given in the ZPSS and are used directly in the Zion review analysis. By calculating the value of the product, (0.15) (CCW)' (EP)', for each of the eight electric power states, and summing them up, one obtains 7.5(-4). Multiplying this number by the initiating event frequency, 0.061, the CDF for this accident sequence was found to be 4.6(-5).

(3) Loss-of-Offsite Power: Loss of Component Cooling Water: Failure to Restore Power in One Hour, SEFC

This accident sequence is essentially identical to that delineated in (2), except for the time at which offsite power is restored. The probability for failure to restore power in one hour (but success prior to four hours) is estimated to be 0.13. By imposing only this change in probability, the CDF for this sequence was quantified, in exactly the same manner, as that described in (2), are found to be 4.0(-5).

(4) Loss-of-Offsite Power: Loss of Component Cooling Water: Failure to Restore Power in Eight Hours, SEFC

This accident sequence is also similar to that discussed above in (2) or (3), except for the timing of offsite power restoration. The plant-damage state being considered is SEFC, which implies success of both the containment fan system and containment spray. The probability for failure to restore offsite power in eight hours was estimated to be 0.1. The success criterion for

the containment fan system is 3-out-of-5 fan coolers operating, which necessitates the availability of power from 2-out-of-3 Unit 1 ac buses. For each of the four electric power states having at least two buses available, therefore, this probability, 0.1, is multiplied by (CCW)', the conditional probability of CCW failure given loss of offsite power and (EP)', the conditional probability of power state, and summed together to yield 1.3(-4). Multiplication of this sum by the loss-of-offsite power initiating frequency, 0.061, yields the CDF for this sequence, 7.9(-6).

(5) Loss-of-Offsite Power: Failure of Component Cooling Water: Failure to Restore Power in Eight Hours, Failure of Containment Fans, SEC

The major difference between this accident sequence and that delineated above in (4) is the inclusion of additional containment fan system failure. Quantification of CDF can be continued from (4) by calculating the product, (0.1) (CCW)' (EP)', for each of the three electric power states in which only one bus is available. These products sum up to 2.9(-4). Since the plant-damage state under consideration is SEC, which implies success of containment spray system, the power state with no power available on any of the three emergency buses is excluded. With no power on any of the emergency buses, the containment spray system automatically fails with probability of 1.0, since power from an emergency bus is required to open the normally-closed MOV CS-006 in the outlet of the diesel-driven containment spray pump. Multiplying 2.9(-4) by the loss of offsite power initiating event frequency, 0.061, yields 1.8(-5) as the CDF for this sequence.

(6) Failure of DC Bus 112, Failure of Auxiliary Feedwater, TEFC

Failure of the dc Bus 112 will not only cause loss of main feedwater and reactor trip, but also render one of the two power operated relief valves (PORVs) inoperable. Since the dc Bus 112 provides control power for the ac Bus 149, ac power to one of the auxiliary feedwater pumps will likewise be lost. With loss of both main and auxiliary feedwater, core heat removal must rely upon "feed and bleed" cooling, which, in this case, cannot succeed because it requires both PORVs to be operational. This accident sequence, therefore, eventually leads to core melt.

The frequency for loss of the dc Bus 112 is given in the ZPSS as 0.28, which was judged to be reasonable by the Zion review. The frequency of auxiliary feedwater system failure, given the unavailability of the ac Bus 149, is estimated to be 2.3(-4) in the Zion review. Since the probabilities for loss of main feedwater, reactor trip, and failure of "feed and bleed" cooling are all unity, the frequency of this sequence can be calculated as 0.28x2.3(-4) = 6.4(-5). By applying a recovery factor to account for possible recovery action by the operators, the sequence CDF was finally estimated to be 7.0(-6). Since the loss of the dc Bus 112 does not interfere with the function of containment fans and sprays, the sequence plant-damage state is TEFC.

(7) Small LOCA: Failure of Recirculation Cooling, SLF

This accident sequence, identified as the foremost contributor to core melt in the ZPSS, is initiated by a small LOCA, followed by failure of recirculation cooling (R-2). The dominant sequence occurs when ac power is available at all three buses. The mean initiating event frequency for small LOCAs

is estimated to be 3.54(-2) in the ZPSS, which is reasonably consistent with the available data. The mean value for the probability of the event R-2 was evaluated as 4.55(-4). Multiplication of these two numbers and the probability of power at all ac buses (-1), together, yields 1.6(-5) as the estimated CDF for this sequence. The ZPSS estimates for the unavailability of high head recirculation (ZPSS, p. 1.5-463) is dominated by the human error of failure to initiate the switchover from injection to recirculation phase and the blockage of the containment sump. Because of lack of data, the Zion review did not evaluate the reasonableness of these largely subjective estimates. Although the Zion review points out some minor computational inconsistency in the ZPSS estimate, it accepts 1.6(-5) as the CDF for this sequence.

(8) Large LOCA: Failure of Recirculation Cooling, ALF

Among the internally initiated accidents, this sequence ranks second in the ZPSS list of leading contributors to core melt. The mean frequency for large LOCA initiating events is presented in the ZPSS as 9.4(-4) per reactor year, which is deemed consistent with the available data. Failure of recirculation cooling (R-1) refers to failure of low-pressure (or low head) recirculation. In the ZPSS, the unavailability of low head recirculation was estimated on the assumption that fan coolers are unavailable and that all three ac buses are available (ZPSS p. 1.5-475). The dominant contributor to this unavailability is the human error of failure to initiate switchover from injection to recirculation. The mean value for R-1 was evaluated in the ZPSS to be 5.19(-3), which, when multiplied by the initiating event frequency, yields the CDF for this sequence, 4.9(-6). The Zion review concurs with this result although it gives some remark concerning minor discrepancy caused by the precision of the ZPSS estimates.

The Zion review also points out that the exclusion of catastrophic reactor vessel rupture in the ZPSS large LOCA analysis is a somewhat nonconservative approach. Depending upon how the data presented in the WASH-1400 are interpreted in terms of percentiles, the Zion review claims that the frequency of vessel rupture could become potentially significant.

(9) Medium LOCA: Failure of Recirculation Cooling, ALF

Owing to the similarity in both the initiating frequency and operational procedures between this sequence and that involving large LOCA with recirculation cooling failure, the analysis and results presented in the ZPSS and discussed in (8) above also apply to this sequence. The comments by the Zion review also remain unchanged.

(10) Loss-of-Offsite Power: Failure of Component Cooling Water: Failure to Recover Offsite Power in Eight Hours: Failure of Containment Sprays and Fan Coolers, SE

This sequence is analogous to that discussed above in (5) except that it also involves failure of containment sprays. Loss of power from ac buses in Unit 1 constitutes the dominant cause for failures of both containment fans and sprays in this sequence. Evaluation of CDF for this sequence can be performed by modifying the computational procedures delineated in (4) and (5). The value of the product, (0.1) (CCW)' (EP)' (unavailability of containment

sprays) (unavailability of fans), is evaluated for each of the eight electric power states and summed together to yield 7.7(-5). Multiplying this number by the initiating event frequency gives 4.7(-6) as the CDF of this sequence.

(11) Large LOCA: Failure of Low-Pressure Injection, AEFC

The large LOCA initiating frequency is 9.4(-4), as stated previously in (8). The mean probability for failure of low-pressure injection system is given in the ZPSS as 1.39(-3), which was reexamined in the Zion review and found to be acceptable. The dominant contributor to this failure is a human error of leaving both MOVs 8812A and 8812B in the low-pressure injection system open after testing and without being detected in the control room. Multiplication of this probability and the initiating frequency together yields 1.31(-6) as the CDF for this sequence.

(12) Loss-of-Offsite Power: Failure of Auxiliary Feedwater: Failure of Feed and Bleed: Failure to Restore Offsite Power in Four Hours, TEFC

This sequence is initiated by loss-of-offsite power, followed by loss-of-auxiliary feedwater and loss of "feed and bleed" capability, with failure to restore power in four hours. The loss of auxiliary feedwater incapacitates steam generator secondary cooling because without offsite power, the main feedwater pumps are tripped and cannot be restored. The loss of "feed and bleed" capability eliminates the remaining option for core heat removal.

As was the case in (2), the frequency of loss-of-offsite power as an initiating event was taken to be 0.061 in the Zion review by including the frequencies of loss-of-offsite power following turbine trip, loss of main feedwater, or reactor trip. This inclusion was made because each of these events could lead to the accident sequence in question. The Zion review also assumes simultaneous loss of offsite power to both Zion units, which is consistent with the ZPSS analysis. The probability of failure to restore offsite power in four hours (but success prior to eight hours) was estimated to be 0.15, as mentioned in (2). The conditional probability for failure of auxiliary feedwater, (AFW)', or failure of feed and bleed, (F&B)', given loss-of-offsite power, is dependent on the state of the vital ac power buses. They were estimated for each of the eight electric power states and are shown in the Zion review (p. 2-33 and p. 2-49).

To evaluate the CDF, the value of the product, (0.15) $(\overline{CCW})'$ (AFW)' (F&B) (EP)', is calculated for each of the eight electric power states and summed together to yield 1.8(-5). The symbols, $(\overline{CCW})'$ and (EP)', denote, respectively, the conditional probability for success of component cooling water and the conditional probability of power state, given loss-of-offsite power. Multiplying this number by the initiating event frequency, 0.061, one obtains 1.1(-6) as the CDF for this sequence.

(13) Loss-of-Offsite Power: Failure of Auxiliary Feedwater: Failure of Feed and Bleed: Failure to Restore Offsite Power in One Hour, TEFC

This sequence is basically identical to that discussed in (12) except that the time for failure to restore offsite power is one hour rather than four hours. Quantification of the CDF can, thus, be proceeded in exactly the same way as that delineated in (12), by only changing 0.15 to 0.13, which

corresponds to the probability for failure to restore offsite power in one hour, as stated previously in (3). This yields 1.0(-6) as the CDF for this sequence.

(14) The Interfacing Systems LOCA, Event V

From the viewpoint of core damage, the interfacing systems LOCA (ISL) is not regarded as a dominant sequence. However, it can lead to release category 2, which, according to the ZPSS estimates, is one of the risk dominating releases. For the Zion plant, the dominant sequence for Event V is the joint failure of two motor-operated valves located in the residual heat removal (RHR) suction line inside the containment. This line is used when the RHR system is in operation during plant shutdown conditions. At the time of RCS startup, these two interlocked motor-operated valves are closed because, otherwise, the RCS repressurization cannot take place. Subsequent failures of both of these valves can then cause the low pressure piping section in the RHR line outside the containment to be inadvertently exposed to high RCS pressure, leading to possible pipe rupture and direct release of radionuclides to the surrounding atmosphere.

The probability for the occurrence of Event V was estimated in the ZPSS by considering logical combinations of the failure modes of these valves including valve disc rupture and valve disc failing open after use. The method and data used in this estimate were critically reexamined by the Zion review, which pointed out several shortcomings of the ZPSS analysis such as the mismatching of the textual description and the model. By employing a newly developed model for the Indian Point plant, an alternative but approximate reevaluation was carried out by the Zion review, yielding a somewhat higher value of 1.6(-7) for the probability of occurrence of Event V. A statistical analysis based on available data, however, indicates that an upper 95% statistical confidence limit on the probability of Event V is 1.4(-7). In view of these results, the Zion review finally drew a conclusion that the ZPSS value of 1.05(-7) is a reasonable point estimate.

(15) Seismic: Loss of All AC Power, SE

This externally initiated sequence occupies the second and the eighth positions in the Zion IDCOR baseline core-damage risk profile and that of the Zion review, respectively. The sequence is of special significance because it has been identified to be the largest contributor to the societal risk among all the sequences listed in the Zion IDCOR baseline risk profile.

In essence, the initiator of this sequence consists of a seismic event severe enough to cause failures of both offsite power and the service water system. Failure of the service water system causes subsequent failure of the emergency diesel generators which rely upon service water for cooling. As a consequence, loss of all ac power arises, followed by failure of the RCP seal cooling and eventually a RCP seal LOCA. Since safety injection and containment heat removal systems require ac power, the loss of RCS inventory cannot be made up, thus leading to core melt with core-damage state, SE. The frequency of this sequence has been evaluated in the ZPSS to be 5.6(-6), which was judged to be reasonable by the Zion review. This frequency represents an insignificant fraction (about 1.6%) of the total CDF estimated by the Zion review, because of the new emergence of several dominating sequences

pertaining to the failure of component cooling water system. In the Zion IDCOR baseline core-damage profile, however, it constitutes roughly 13% of the total CDF , as stated previously. The fact that the plant-damage state corresponding to this sequence is SE, meaning early core melt without containment cooling, makes this sequence a potentially important contributor to risk and release consequences.

A.1.3 SARP Rebaseline Estimate of Zion Core-Damage Frequency

The rebaselining study on Zion was limited essentially to reexamination and updating of the models and core-damage sequences presented in the Zion review. One major accomplishment of the study was the evaluation of the potential impact of certain issues on the dominant sequence CDF and plant damage states. These issues include both plant operation changes and some generic PRA issues which have emerged since the time of the Zion review. In addition to the Zion review, the study also made reference to the Zion Probabilistic Safety Study (ZPSS), and the NRR Staff Report[4] on Zion (August 1, 1985). The letter (of July 9, 1984) sent from Dave Kunsman of SNL to Scott Newberry of NRR was also used as a valuable source of information because it presents the new sequence equations and estimates for the Zion review sequences influenced by the imposition of new success criteria for the component cooling water system (CCWS) and service water system (SWS). The letter, in effect, represents a forerunner to a limited rebaselining of the Zion review, since it constitutes the quantitative bases for the NRR Staff Report and the SARP Study.[2]

In the Zion limited rebaselining study, the scheme for labeling the sequences and defining the plant damage states follows exactly that developed in the ZPSS (and adopted by the Zion review). The sequences are simply identified by their ranking among the dominant sequences. Also, only one plant damage state is assigned to each sequence. The plant-damage state is characterized by using a four letter set, with the first letter describing the initiator (T≡transient, A≡Medium or Large LOCA, S≡Small LOCA), the second letter the phase of reactor cooling (injection or recirculation) failure during which core melt occurs (E≡Early core melt with injection failure, L≡Late core melt with recirculation failure) and the last two letters denoting respectively successful operation of the containment fan coolers and spray system (C≡successful containment spray injection or recirculation, F≡successful containment fan coolers).

The rebaseline Zion dominant core damage sequences and the corresponding plant-damage states are shown in Table A.4, along with those developed by the Zion review and IDCOR Baseline for ready comparison. Since no attempt was made to identify any new dominant sequences, the rebaselining effort only brought about shifting and rearrangement of the ranking of the sequences established by the Zion review. In particular, the following points can be noted by comparing the two sets of core-damage profiles.

1. The seismic event with loss of all ac power, the eighth sequence in the Zion review, is removed from the rebaselining list, apparently because it is an externally initiated event that is beyond the scope of the study.

2. There are five sequences whose CDF remain unchanged from those estimated by the Zion review. Falling under this category are the small, medium, or large LOCAs followed by failure of recirculation cooling (sequences 2, 3, and 4), large LOCA followed by failure of low-pressure injection (sequence 6) and the ISL (sequence 14).

3. For all but one (sequence 5) of the remaining sequences, the CDF re-evaluated by the rebaselining are significantly reduced. The increase in the CDF for sequence 5, by roughly a factor of two, is chiefly attributable to the substantially larger probability estimated for the failure to restore ac power in one hour (but success in four hours).

4. As will be discussed in more detail later, the three sequences (sequences 9, 12, and 16) which did not appear in the Zion review represent portions of certain Zion review sequences newly created as a result of including loss of SWS as a contributor to core damage.

5. Owing to the credit newly given to refilling the refueling water storage tank (RWST), which enables long term use of the containment sprays, the plant-damage state for sequence 2 is reclassified as SLFC instead of SLF. By the same reason, the plant-damage states for sequences 3 and 4 are both changed from ALF to ALFC, implying successful operation of the containment spray system as well as the containment fans. Consideration of this new credit, however, did not affect the CDF of these sequences.

Focusing attention on the top six dominant sequences, it is of interest to observe that, with the exception of sequence 1, the loss of CCWS following the loss of offsite power no longer occupies leading positions in the core damage list. Instead, these positions are taken by the LOCAs of various sizes, followed by failure of recirculation cooling or low-pressure injection, a trend which prevailed in the ZPSS core-damage profile. The relative importance of CCWS failure is diminished primarily by virtue of the relaxation in the success criterion for the CCWS. In the Zion review, the success criteria for the CCWS and the SWS were assumed to be functioning of 2-of-5 CCWS pumps and 2-of-6 SWS pumps. These criteria were eased somewhat in the rebaselining study, based on the concurrence between NRR staff and Commonwealth Edison, to 1-of-5 CCWS pumps, while maintaining the same 2-of-6 SWS pumps success criterion. It should not be overlooked, however, that this less stringent CCWS success criterion increases the importance of SWS with respect to CCWS. The relative importance of SWS can have significant impact on the plant damage state since its failure will cause failure of the containment fans and spray pumps, while CCWS failure will not.

The rebaselining study, therefore, took into consideration the SWS failure, which was not included in the Zion review models. The sequence equations presented in the Kunsman's letter, which made no distinction between loss of CCWS and loss of SWS, were requantified, factoring out the SWS terms from the CCWS terms to determine the impact of SWS failure on plant-damage states. The rebaseline models also incorporate pumps and diesel generator common mode failure events, which were not considered in the Kunsman's analysis.

The essential details on the quantification of CCWS and SWS unavailabilities based on the new success criteria are presented in Section IV.4.1 of the rebaseline analysis report.[2] Table IV.4.1 of the same report tabulates the calculated unavailabilities of these two systems as a function of the eight electric power states.

In addition to the CCWS and SWS pump success criteria discussed above, several other important issues were considered in the rebaseline analysis, a concise summary of which is given in the following.

(a) Common Mode Failures

The common mode failures among CCWS, SWS, and Unit 2 diesel generator, which were not included in the models of ZPSS, the Zion review and Kunsman's letter, were taken into account in the rebaseline analysis (1) to enhance the consistency between the Zion NUREG-1150 work and other NUREG-1150 analyses, and (2) to take advantage of the detailed nature of the equations in Kunsman's letter, which permits easy application of the NUREG-1150 common mode analysis to the Zion models. The diesel generator common mode event, however, was modeled only for the two diesel generators dedicated to Unit 2, and not modeled across Units 1 and 2.

(b) Restoration of AC Power

The probability of failure to restore offsite power within a given time span was adjusted in the sequence models to reflect the changes in the generic power recovery models which have taken place since the time of the Zion review. Additionally, failure to restore a diesel generator was also included in the sequence models. The failure probabilities for restoring offsite power and diesel generators are taken from the ASEP generic data base.

(c) Feed and Bleed

The procedures for feed and bleed are said to be in place at Zion. In the Zion review, one of the primary requisites to successful feed and bleed cooling, in case of complete loss of feedwater and AFW, is the functioning of two pressurizer PORVs. On the ground that the shutoff head of the two charging pumps (2670 psig) is sufficiently higher than the pressure set point for the safety relief valves (2435 psig), the rebaseline analysis assumed that only one PORV needs to be operational for successful operation of feed and bleed.

(d) RWST Refill

Procedures are said to be in place at Zion for replenishing the RWST should such a need occur. The rebaseline analysis, therefore, gives credit to the RWST refill, which does not have any impact on the CDF. It can, however, alter the plant-damage states of certain sequences, because, if RWST is replenished, the containment spray injection could be continued as a substitute for failed containment recirculation cooling.

(e) Reactor Coolant Pump Seal LOCA

The crucial question of how large and how quickly a seal LOCA will develop upon loss of CCW cooling to the RCP seals remains an unresolved issue. The Zion review assumed that a 300 gpm leak per pump would develop within one hour if CCWS flow to the RCP seal is lost, a postulation deemed potentially conservative by the NRR staff. The seal LOCA model used in the rebaseline analysis is a time dependent Weibull distributed random variable with a 5% confidence limit at one hour and 95% confidence limit at ten hours. It implies that, after the CCWS cooling is lost, there is a 0.05 probability that the seals will have failed within one hour, and 0.95 probability that the seal will have failed within ten hours. The rebaseline sequence models also assumed that all RCP seals fail concurrently, with a leakage rate of 450 gpm per pump, for a total seal LOCA of 1800 gpm. It is further assumed that, once a seal LOCA occurs, core uncovery will not take place for a period of one hour even without injection to the primary system. For loss of offsite power transients resulting in failure of high-pressure injection, this allows for an extra hour for recovery of ac power.

(f) Manual Switchover to Recirculation Cooling

In the event of a LOCA (such as sequences 2, 3, and 4 in Table A.4), or under certain transient conditions, the operators must manually switch the emergency core cooling system (ECCS) over from injection phase to recirculation phase, when the RWST reaches a low level. Human errors involved in this manual switchover are the leading contributors to the CDF of these sequences. Since no new information suggesting improved reliability in this human action was received, the rebaseline analysis made no changes in the recirculation cooling models.

(g) Recovery of CCWS Pipe Ruptures

In both the Zion Review and the rebaseline study, the most dominating core-damage sequence was identified to be CCWS failure, causing failure of all charging and safety injection pumps and the development of RCP seal LOCA, as shown in Table A.4. Because of pipe rupture the CCW system failure plays a decisive role in the quantification of the CDF for this sequence. In the Zion review, the frequency of CCWS pipe rupture was estimated based on the mean pipe rupture data (per hour per section) given in the ZPSS, which originated from WASH-1400. In computing the CDF of this sequence, however, credit was given to recovery of CCW pipe because of the availability of procedures in Zion to isolate pipe leaks in the CCWS. There are, however, some uncertainty about the credibility of isolating such leaks in a timely manner. Furthermore, no data are available for a statistical analysis of this recovery action. Notwithstanding, the rebaseline analysis retains the assumptions made in the Zion Review.

(h) RHR System Check Valve Testing

It is required by the NRC Confirmatory Order of 1980 that the RHR check valve disks be tested for integrity at every refueling shutdown and cold shutdown. Without such testing, the frequency of an ISL would increase to

1.0E-6,* an order of magnitude larger than that predicted by both the ZPSS and Zion review. Since compliance with the Confirmatory Order by the Zion station has been confirmed, the ISL analysis of the Zion review was considered adequate.

(i) Diesel-Driven Containment Spray Pump

Owing to its need for SWS cooling water, the one diesel-driven containment spray pump in Zion still has an indirect dependency on ac power. According to the NRR Staff Report, modifying one or more of the containment spray pumps to be entirely ac independent could have significant impact on reducing the risk after core melt. Since no such modifications are currently contemplated by the utility, the impact of this issue on plant damage state frequencies was investigated only as a sensitivity study in the rebaseline analysis.

To facilitate understanding of the methodology used in the Zion rebaseline analysis for developing the dominant core-damage sequences shown in Table A.4, a brief discussion of each sequence is given below in order of sequence dominance. Emphasis is placed upon describing the analytical methods and logical basis of rebaseline sequence frequency quantification in relation to the Zion review. The Zion review sequence frequencies are shown inside parentheses.

(1) Sequence 1. CCWS failure, causing loss of all charging and SI pumps, loss of RCP seal cooling resulting in RCP seal LOCA. 1.2E-4 (2.0E-4).

This sequence, identified to be the most dominant by both the Zion review and the rebaseline analysis, is essentially dominated by the frequency of pipe rupture event in the CCWS. The sequence frequency estimated by the Zion Review is reduced by roughly a factor of two by virtue of the less stringent success criterion for the CCWS (1-of-5 pumps). The frequency of CCW pipe rupture leading to core melt, 1.2E-4, assessed by the Zion review upon considering 50% recovery factor, is retained in the rebaseline analysis. The frequency of CCWS failure because of causes other than the pipe rupture, 7.1E-4, is considered recoverable by successfully starting one of two standby pumps (as opposed to 2-of-2 standby pumps in Zion review). By using the same unavailability data for the standby pump (because of maintenance or failure to start) as those shown in the Zion Review, the probability of failure to restore CCWS was found to be 7.7E-3. The rebaseline sequence frequency is, therefore, 1.2E-4+(7.1E-4)(7.7E-3) = 1.2E-4. It is, thus, demonstrated that this sequence frequency is almost totally dominated by the frequency of CCWS pipe rupture. The lack of reliable data for CCWS pipe rupture, however, arouses a question regarding the suitability of the pipe rupture frequency, which derives its origin from the WASH-1400, used in this quantification.

The WASH-1400 pipe rupture data are more representative of rupture of large pipes exposed to the environment of the primary system, which is subjected to relatively high temperature and high pressure. Whether or not such data are applicable to the rupture of CCWS, which is normally under lower pressure and significantly lower temperature, is certainly a debatable issue.

*1.0E-6 = 1.0×10^{-6}.

In view of the dominating nature of CCWS pipe rupture frequency and the unavailability of applicable data, a parametric study was performed, in the rebaseline analysis, to study its effect on the CDF of this sequence. Based on a revised estimate of pipe ruptures for low-pressure piping, the SARRP study performed by Brookhaven National Laboratory (BNL) has reduced the estimated CDF to 5.5×10^{-6} per reactor year.

(2) Sequence 2. Small LOCA, followed by failure of recirculation cooling. 1.6E-5 (1.6E-5).

The only modification made to this sequence, the fifth ranked sequence in the Zion review, is to grant credit for refilling the RWST, which would allow for long-term use of the containment spray injection system (CSIS). The plant damage state is, thus, reclassified from SLF to SLFC, with no change made in the sequence frequency. Although availability of the RWST is postulated, there are enough factors favoring this seemingly nonconservative assumption. Since the failure occurs during recirculation phase, the operators would have sufficient time to decide on the need for refilling the RWST and to take such actions if necessary. Moreover, this sequence does not involve any degraded power situations, which could adversely affect the success of refilling the RWST.

(3) Sequence 3. Large LOCA, followed by failure of recirculation cooling. 4.9E-6 (4.9E-6).

Just as for sequence 2 discussed above, the only change from the Zion Review analysis is the reclassification of the plant damage state from ALF to ALFC, by awarding credit to refilling the RWST. The sequence frequency remains unchanged from the Zion review analysis.

(4) Sequence 4. Medium LOCA, followed by failure of recirculation cooling. 4.9E-6 (4.9E-6).

The comments given above on sequence 3 also apply to this sequence.

(5) Sequence 5. Loss-of-offsite power, failure of auxiliary feedwater (AFW), failure of feed and bleed, failure to restore ac power in one hour (but recovery prior to four hours). 2.1E-6 (1.0E-6).

As mentioned earlier, this is the only sequence which involves an increase in the sequence CDF in comparison with the Zion review. The factors contributing to this increase are explained in the following. Three principal modifications are made to the Zion review analysis for this sequence, including recalculations of the probabilities for (a) failure to restore ac power in one hour (recovery prior to four hours), (b) failure to operate SWS and CCWS based on the new success criteria (1-of-5 CCWS pumps and 2-of-6 SWS pumps), and (c) failure of feed and bleed based on the new success criterion (1-of-2 PORVs).

Of these, the probability for (a) was reevaluated to be 0.23, a value substantially larger than that of 0.13 used in the Zion review. As stated previously, the probabilities for (b) have been calculated as a function of the eight power states, and are tabulated in Table IV.4.1 of the rebaseline

analysis report.[2] Since the plant-damage state for this sequence is TEFC, the main quantities of interest here are the probabilities of successful operation of SWS and CCWS, the complements of the unavailabilities of SWS and CCWS. For those four power states where there are at least two emergency ac buses available, these complements are all practically equal to 1.0, the same as those found in the Zion review. For the three power states which have only one emergency power bus available, these complements also take the value of 1.0, which is slightly larger than the 0.97 used in the Zion review. For the power state with no bus available, the complement was found to be 0.97 rather than the 0.83 used in the Zion review. It must be remarked that, for the Zion Review, these complements only reflect the unavailabilities of CCWS, since the failure of SWS was not considered.

The probabilities for the failure of feed and bleed (c) above, based on the new and more relaxed success criterion of 1-of-2 PORVs, are obtained by adding together the probabilities of three distinctive events, namely (i) all charging and SI pumps fail, (ii) one PORV and both charging pumps fail, and (iii) both PORVs fail. Occurrence of any of these events constitutes failure of feed and bleed based on the new success criterion. The probabilities for the event (i) have been evaluated as a function of the eight power states in the Zion review (p. 2.31), and are adopted directly. The probability for failure to open 1-of-2 PORVs was estimated to be 2.9E-3, based on the data shown in the Zion review (page 2.32). The PORV block valves are assumed to be normally open. The probabilities for the failure of both charging pumps, as a function of the eight power states, are also available from the Zion review (page 2.31). When they are multiplied by 2.9E-3, the probabilities for event (ii) can be obtained. The probabilities for event (iii) were estimated to be 1.3E-4, also based on the data shown in Zion review (page 2.32).

To calculate the CDF for this sequence, the product, (0.23) (probability for successful operation of SWS, CCWS) (failure probability of feed and bleed) (AFW failure probability) (probability of power states), is evaluated for each of the eight power states and summed together. It should be reiterated that only the first three probabilities appearing in this product have been reevaluated from those used in the Zion review analysis. The summation, after multiplying by the initiator frequency, 0.061 per year, yields 2.1E-6 as the CDF for this sequence. It was determined that, the increase in the sequence frequency, as compared to the Zion review, is primarily because of the larger probability found for failure to restore ac power within one hour. The new success criterion for feed and bleed was found to have little impact on the sequence CDF, because it is dominated by loss of ac power to the charging and SI pumps.

(6) <u>Sequence 6</u>. Large LOCA, followed by failure of low-pressure injection. 1.4E-6 (1.4E-6).

No modifications or changes were made on the Zion review sequence model for this core damage sequence, which was sequence 12 in the Zion review.

(7) <u>Sequence 7</u>. Loss-of-offsite power, followed by AFWS failure, failure of feed and bleed, failure to restore ac power within four hours (but successful restoration by eight hours). 4.6E-7 (1.1E-6).

This sequence is exactly identical to sequence 5 except for the timing for restoring the ac power.

As a result, the discussions presented above in (5) with regard to recalculating the failure probability of feed and bleed or the unavailabilities of CCWS and SWS, based on the new success criteria, are directly applicable here. The only difference lies in the probability of failure to restore ac power, which was found to be 0.05 for this particular time span. It is noteworthy that this value correspond to one-third of that used in the Zion review. Calculation of the sequence CDF can be proceeded by following the description given in the last paragraph of (5) above, with only replacing 0.23 by 0.05. It was found to be 4.6E-7/yr, less than one-half of that estimated by the Zion review (sequence 13).

(8) <u>Sequence 8</u>. Loss-of-offsite power, loss of either CCWS or SWS, failure to restore ac power in one hour (but successful restoration of ac power by four hours). 3.2E-7 (4.0E-5).

The SWS failure considered here refers to the recoverable SWS failures such as those caused by diesel generator failure. Most of the failures in the SWS model are recoverable upon successful restoration of ac power. The nonrecoverable SWS failures are separated out and defined as a new sequence, sequence 9. No counterpart of sequence 9 exists in the Zion review, since SWS failure was not considered in the Zion review.

The probability for failure to restore ac power was determined to be 0.23 for the power state "All" (implying ac power available at all buses), and 0.26 for the remaining seven power states because of the additional one hour the NUREG-1150 RCP seal LOCA model allows for ac restoration over that in the Zion review models. In the Zion review, 0.13 was used for all the eight power states.

For the power state "All," an RCP seal LOCA is assumed to take place with probability 1.0, since loss of CCWS is almost certain to occur because of common mode (nonrecoverable) failures. For the remaining seven power states, however, the probability for a seal LOCA to occur at one hour was taken to be 0.05, based on the NUREG-1150 RCP seal LOCA model. For the unavailabilities of both SWS and CCWS, the beta factors in Ref. 6 for the common mode event probabilities, shown in Table IV.4.1 of the rebaseline report, are used.

The sequence CDF was computed by first evaluating the product, (failure probability to restore ac by one hour) (probability for loss of CCWS or SWS) (power state probability) (probability of RCP seal LOCA at one hour), for each of the eight power states and summed up to obtain 5.2E-6. Multiplying this by the frequency of loss of offsite power, 6.1E-2/yr, gives 3.2E-7/yr as the CDF of this sequence.

(9) <u>Sequence 9</u>. Same as sequence 8, except this represents the SWS common mode failure portion of the Zion review sequence No. 3. 3.0E-7 (none in the Zion review).

As stated previously in (8), this new sequence is created as a result of separating out the nonrecoverable SWS failures from the CCWS failures in sequence 8. Nonrecoverable failures refer to SWS failures which can not be recovered upon restoration of ac power. This sequence is almost totally dominated by common mode failure of the SWS pumps. Since SWS failure is not recoverable, the CCWS and the injection pumps will not be operable, and an RCP seal LOCA will develop with probability 1.0. The plant-damage state of this sequence is SE, since both the containment sprays and containment fans depend on SWS. The probability for failure to restore ac power within one hour (but, success by four hours) was found to be 0.23 for all the power states. Also, the unavailability of the SWS not caused by diesel generator failure was determined to be 2.2E-5. Evaluating the product, (0.23) (2.2E-5) (power state probability) (1.0), for each of the eight states, and summing them up yields 4.9E-6. Multiplying this by the frequency of loss of offsite power, 6.1E-2/yr, gives 3.0E-7/yr as the CDF for this sequence.

(10) <u>Sequence 10</u>. Loss-of-offsite power, loss of CCWS or SWS, failure to restore ac power within eight hours, failure of containment sprays and fan coolers. 2.0E-7 (4.7E-6).

This sequence is similar to sequence 8, except that ac power is not restored in eight hours. Both the Zion review and ZPSS gave no credit for containment systems if they could not be restored within eight hours.

Contributions to the sequence CDF were evaluated separately for failures of CCWS and SWS. As in the Zion review, only the four highly degraded power states (with only one bus or none available) were considered in estimating the contribution because of CCWS failure. The probability for failure to restore ac power within eight hours was found to be 0.02, a smaller value compared to the 0.1 used in the Zion review. For those highly degraded power states, the unavailability of containment fans is 1.0. To calculate the CCWS contribution, the product, (failure probability to restore ac by eight hours) (unavailability of CCWS) (unavailability of containment sprays) (power state probability), is evaluated for each of the four degraded power states and summed together to obtain 8.4E-7.

The SWS contributions are assessed by considering all the eight power states, since loss of SWS causes failure of both containment sprays and fans with probability 1.0, given loss-of-offsite power, because of the eventual station blackout upon loss of diesel generator cooling.

The product, (failure probability to restore ac in eight hours) (unavailabilities of SWS) (power state probability), is evaluated for each of the eight power states and added together to yield 2.4E-6. Multiplying (8.4E-7 + 2.4E-6) by the frequency of loss of offsite power, (6.1E-2), the CDF of this sequence is obtained as 2.0E-7.

(11) Sequence 11. Loss-of-offsite power, loss of CCWS or SWS, failure to restore ac power within four hours (successful restoration by eight hours). 1.5E-7 (4.6E-5).

This sequence differs from sequence 8 only in that the time span over which recovery of offsite power and the probability of an RCP seal LOCA are modeled is prolonged. An RCP seal LOCA is considered to occur, in this sequence, at four hours, and ac power is successfully restored within four to eight hours for power state,"All," and five to eight hours for the remaining degraded power states, taking into account the additional one hour allotted for restoring ac power after the seal LOCA occurs. The discussions given above in (8) regarding the RCP seal LOCA model and restoration of offsite power are also applicable here. The nonrecoverable SWS portion of sequence 11 is calculated in sequence 16. The probabilities for failure to recover ac power within four to eight hours and five to eight hours were estimated to be 0.05 and 0.02, respectively. For the degraded power states, the probability for RCP seal LOCA to occur at four hours was evaluated to be 0.47 based on the NUREG-1150 RCP seal LOCA model. With only these two changes, the sequence CDF can be calculated, following the description given for sequence 8, to be 1.5E-7/yr.

(12) Sequence 12. Loss-of-offsite power, failure of SWS, failure to restore ac power within eight hours. This sequence represents the SWS failure portion of the rebaselined Zion review sequences No. 4 and No. 6. 1.5E-7 (no corresponding sequence in Zion review).

This sequence, newly created by the inclusion of SWS failure into the sequence model, should be considered as an offshoot of sequences 13 and 17, in which SWS failure contributes to core melt. These two sequences had to be redefined into three sequences, because the ensuing plant-damage states depend upon whether CCWS or SWS contributed to the sequences. This sequence is newly formed by separating out those portions of sequences 13 and 17 which are attributable to SWS failure. No counterpart of this sequence can be found in the Zion review, since SWS was not included in the sequence models. For this sequence, all SWS failures are unrecoverable because ac power is not restored within eight hours, leading to complete loss of containment sprays and fans. The plant damage state is, accordingly, SE, as compared to sequence 13 (SEC) and sequence 17 (SEFC), both of which involve CCWS rather than SWS failures. The probability for failure to restore ac power within eight hours was estimated to be 0.02.

The sequence frequency quantification can be carried out by evaluating the product, (0.02) (unavailability of SWS) (power state probability), for each of the eight power states, and summing them up to obtain 2.4E-6. Multiplying this by the frequency of loss of offsite power, 0.061/yr, gives 1.5E-7/yr as the CDF of this sequence.

(13) Sequence 13. Same as sequence 12 above, except this is the CCW portion of the Zion review sequence No. 4, and containment fans fail. 1.0E-7 (1.8E-5).

This sequence involves a set of events similar to those in sequence 12, except that containment fan failure is explicitly included here. It

corresponds to sequence 4 in the Zion review, and results in plant-damage state SEC, with loss of the containment fans because of degraded power states. The dominant cause of fan failure here is loss of 2-of-3 ac buses at Unit 1 because of random failures in the diesel generators. As mentioned earlier, the success criterion for the containment fan coolers is at least 3-of-5 fans. These fan coolers will fail electrically only for degraded power states with one bus (147, 148, or 149) or no bus available. Since failure of all ac power at Unit 1 will also result in loss of containment sprays, which is not compatible with the plant-damage state SEC, the power state of "no bus available" is excluded. The probability for failure to restore ac power within eight hours is 0.02, as compared to the value of 0.1 used in the Zion review. The unavailabilities of CCW are those reevaluated based on the new success criterion for the CCWS.

For those three degraded power states with only one bus (147, 148, or 149) available, the product, (0.02) (unavailability of CCW) (power state probability), is evaluated and summed up to obtain 1.7E-6. The sequence CDF is, therefore, (6.1E-2/yr) (1.7E-6) = 1.0E-7/yr.

(14) <u>Sequence 14</u>. Interfacing systems LOCA. 1.05E-7 (1.05E-7).

For this sequence, no change was made on the model in the Zion review.

(15) <u>Sequence 15</u>. Failure of dc Bus 112, causing loss of one PORV and loss of ac Bus 148, failure of AFWS. 5.0E-8 (7.0E-6)

The only change made on the Zion review model, for this sequence, is the reevaluation of the probability for failure of feed and bleed based on the new success criterion of 1-of-2 PORVs. It was calculated by adding the probability for operator error to open one PORV (=1.3E-4), and the probability for failure of PORV to open on demand (=1.4E-3). The initiator frequency (0.28/yr), the recovery of initiator (0.1), and the conditional failure probability of AFWS, given the loss of ac Bus 148 (=2.3E-4), are all taken directly from the Zion review. The sequence CDF is, therefore, (0.28/yr) (0.1) (2.3E-4) (1.4E-3 + 1.3E-4) = 5.0E-8/yr.

(16) <u>Sequence 16</u>. Same as sequence 11, except this represents the SWS common mode failure portion of the Zion review sequence No. 2. 4.8E-8 (none in the Zion review).

This sequence is analogous to sequence 11 except that it is caused by nonrecoverable failure of the SWS, which is dominated by common mode failure. As mentioned earlier, SWS failure becomes important relative to CCWS failure because of the new success criterion for CCWS. The nonrecoverable SWS failures, which will result in damage state SE, therefore, must be separated from nonrecoverable SWS failures and CCWS faults. Since the SWS can not be recovered, the CCWS and injection pumps will not be operable, leading to an RCP seal LOCA with probability 1.0. The containment sprays and fans also fail because of the nonrecoverable SWS failure, thus giving rise to plant damage state, SE. As mentioned in (9) above, the unavailability of SWS not caused by diesel generator failure is 2.2E-5. Also, the probability for failure to restore ac power within four hours (success by eight hours) was determined to be 0.06.

The product, (0.06) (2.2E-5) (power state probability) (1.0), is evaluated for each of the eight power states and summed up to give 7.8E-7. Multiplying this by the frequency of loss-of-offsite power, 6.1E-2/yr, yields 4.8E-8/yr as the CDF of this sequence.

(17) Sequence 17. Loss-of-offsite power, CCWS failure, failure to recover offsite power or diesel generators in eight hours. 3.7E-8 (8.0E-6).

This sequence is essentially equivalent to sequence 6 in the Zion review and is similar to sequence 8 in that it involves a set of events leading to core damage because of an RCP seal LOCA. Unlike sequence 8, however, none of the ac power failures are restored within eight hours after initiation of the sequence. Since the plant-damage state is defined to be SEFC, meaning successful operation of both containment fans and sprays, only the power states resulting in SEFC for this set of events are included in the analysis. These power states correspond to emergency power available at all buses or at two buses (147-148, 147-149, 148-149). The probability of an RCP seal LOCA is 1.0, as in the Zion review, whereas that for failure to restore ac power within eight hours was taken to be 0.02 instead of 0.1. The unavailabilities of CCWS are those based on the new CCWS success criterion.

For those four power states, the product, (0.02) (unavailability of CCWS) (power state probability), is evaluated and summed up to give 6.0E-7. Multiplying this by the frequency of loss-of-offsite power, 6.1E-2/yr, yields 3.7E-8/yr as the sequence CDF.

A.1.4 Comparison Between IDCOR and SARP

Table A.5 compares dominant core-damage sequences for Zion between the SARP rebaseline and IDCOR baseline studies. Although the IDCOR baseline risk profile was developed by updating the ZPSS, the IDCOR baseline core-damage sequences and their frequencies are practically identical with those of the ZPSS. The SARP effort for Zion is a limited rebaselining of the models and accident sequences addressed in the Zion review. Thus, the results in the SARP rebaseline study would represent an update of the existing dominant sequences in the Zion review, and no attempt was made to identify any new dominant sequences.

A major difference between the two studies is the loss of component cooling water which may lead to a RCP seal LOCA sequence under high-pressure injection failure. This was identified in the SARP rebaseline study as the most dominant contributor to core damage, which was not, however, considered in the IDCOR baseline study. Minor differences between the two studies are because of the new success criteria used for the CCWS, SWS, and pressurizer PORVs, resulting in slightly different sequence definitions and associated sequence frequencies.

A.1.5 Comparison of Zion Core-Damage Frequency With Other Plants

A comparison of several recent studies of other PWR plants with large-volume containments is given in Table A.6. Note that the estimated CDF for most of the plants is quite close to Zion and that most of the sequence types found at Zion also are found at the other plants although the individual

contribution of each sequence type varies substantially. A substantial fraction of the CDF for each plant is primarily caused by support system failures. Only Oconee exhibits a unique sequence type (internal floods). The Oconee study also identified instrument air as a key support system for which the effects of failure are difficult to quantify but are estimated to be a significant contributor to many core-melt sequences.

A.2 Core-Meltdown Phenomena and Containment Response

This section contains a description of Zion containment performance and a list of available Zion plant accident analyses (Section A.2.1). A comparison of available results from IDCOR and SARP analyses for containment response, radioactivity release and consequences is given in Section A.2.2. The discussion of additional accidents analyzed by NRC contractors for this plant is in Section A.3. The differences between IDCOR and SARP analyses identified in Section A.2.2 are discussed in Section A.3.1. Model differences and important emergency system features, which could diminish the consequences of analyzed accidents, are also discussed. The offsite consequences are discussed in Section A.4.

A.2.1 Containment Performance

The Zion containment is in the shape of a cylinder with a shallow, domed roof, and a flat foundation slab. The pressure at which failure would be expected to occur is a key determinant in the likelihood and timing of containment failure during pressure transients that result from the generation of steam or combustion of hydrogen. The pressure at containment failure can also influence the dispersion of fission products released to the environment. The PRA for Zion Nuclear Power Station determined the realistic containment building ultimate pressure capacity to be about 150 psia (lower bound).

The containment fan cooler spray systems provide redundant and diverse containment heat removal capability for Zion. The containment fan cooler system is designed to remove heat from the containment building during both normal operation and in the event of a design basis accident. The containment fan cooler units are an engineered safeguard system. Five fan coolers are provided for each containment. Each cooler is rated at one-third the required capacity for design basis accident conditions.

The containment spray system, on the other hand, is designed to limit the pressure in the containment atmosphere to levels below the containment design pressure and to reduce the radiological releases to the 10CFR100 limits. Three completely redundant containment spray system trains are provided for each unit, with each system rated at 100 percent capacity for design basis accident conditions. One of the spray trains has a diesel engine driven spray pump for added diversity. All three containment spray pumps take suction from the RWST and discharge into the spray rings located around the inside of the containment dome. Should spray be required during the recirculation phase of the accident, two of the three spray subsystems can be supplied with water from the containment sump via the RHR pumps which deliver water to the discharge lines of the two motor operated spray pumps. Spray pump operation is therefore not necessary during the recirculation phase. Both motor-driven pumps and all motor operated valves can be supplied with power from the

emergency diesel generators in the event of a loss-of-off-site power. Failure of a single diesel or emergency bus will affect one subsystem only.

A.2.1.1 List of Analyzed Accidents

The Zion Nuclear Power Station accidents were analyzed by SNL[3] and BNL[7] in the review of Zion PRA[5] study, by Battelle Columbus Laboratories (BCL) in BMI-2104[9] and a recent study[10] in support of NUREG-1150, and very recently by BNL's rebaselining study.[11] Previously the Zion plant accidents were analyzed by the NRC in NUREG-0880,[12] by BNL in NUREG/CR-2228[13] and reviewed by the NRC staff in Ref. 4. The industry Zion study is reported in Volume 23.1[14] and Volume 21.1[1] of the IDCOR report and in Zion PRA study (ZPSS).[5]

Table A.7 gives a list of analyzed accidents. The status of plant damage is given by two to four letter symbols (e.g., SLFC = Small LOCA, Late containment rupture, Fans operating, Containment Sprays operating, see explanation to Table A.2). The other columns in Table A.7 are defined by symbols from WASH-1400.[15]

A.2.1.2 Containment Failure Probability

The probability for early containment failure from the SARRP study[11] (Figure 3.5 of Ref. 11) is plotted in Figure A.2 along with mean, median values and uncertainty ranges represented by the 5th and 95th percentiles. As shown, the uncertainty range is from 1×10^{-3} to 0.17. The median value is 0.01 and the mean value is 0.04

The results were compared with other studies in Table A.8. The Surry study has a mean value of 0.2 which is about a factor of five higher than that of 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 of the Zion containment showed a similar result to that of the IDCOR analysis. The mean probabilities of isolation failure were .04 in the SARRP study and 5.0×10^{-3} in the IDCOR analysis, respectively.

A.2.2 Comparison of SARP and IDCOR Results

The results of SARP and IDCOR calculations for the primary system and containment response are summarized in Table A.9. The containment pressure response for compared accidents are given in Figures A.3 through A.13. The main features from this comparison are given as follows:

The "Seal LOCA" Accidents (Sequences 1 through 4 in Table A.9)

The "seal LOCA" accidents are either TMLB' type as in Refs. 1 and 14 or S_2D type as in Ref. 10. They are both characterized by small LOCA from pump seals with failure of ECCS cooling injection (D) and failure of containment spray injection (C), which may include failure of spray recirculation system (F). The same failure events occur with TMLB' accident seal LOCA (failure of all ECC systems because of loss of site power).

In the TMLB' case the seal LOCA occurs 45 minutes after loss of site power; in the S_2D case a seal LOCA occurs as an initiating event, at t = 0,

followed by failure of ECC injection and spray injection and recirculation system failure. The S_2D C_{ir} F_{ir} 1 case assumes early containment failure (at 2.22 hrs) because of hydrogen burn overpressure (γ case), the S_2D C_{ir} F_{ir} 2 case assumes late containment failure (at 15 hrs).

Figure A.3 from Ref. 10 shows for S_2D C_{ir} (SARP) containment pressure increasing from 14.7 psia to 149 psi almost linearly during 24 hours, when the containment is assumed to rupture. At the RPV failure one can see (Fig. A.3), a pressure spike with another spike at 2.7 hours (53 psi) because of hydrogen burning. Similarly, for the TMLB', seal LOCA case (IDCOR), given in Figure A.4 from Ref. 14, one can see a steep pressure rise at core uncovery (1.57 hrs) and another steep rise with a spike at RPV failure (3.8 hrs). Subsequently, the pressure rises slowly to 149 psi, when the containment ruptures at 32 hours and very slowly (because of the assumption a small hole in containment) loses pressure (IDCOR case). The containment rupture in SARP case (see Fig. A.3), occurs faster (at 24 hrs) and the containment pressure drops suddenly, because of an assumed large hole in containment. The pressure in containment for SARP S_2D C_{ir} F_{ir} 2 case behaves rather similarly as in SARP S_2D C_{ir} case, except the containment ruptures faster, at 17.93 hours (see Fig. A.5). On the other hand for the SARP S_2D C_{ir} F_{ir}2 case the containment ruptures because of a pressure spike caused by hydrogen burn at 2.22 hours (see Fig. A.6). No such spike is ever observed in the IDCOR calculations. The IDCOR TMLB'-β, seal LOCA case behaves similarly to the TMLB'-δ case, except the containment is assumed to be bypassed, with times of core melt and RPV failure about the same, as for IDCOR TMLB'-δ seal LOCA case.

<u>The TMLB' (no LOCA) cases</u> (Sequences 5 through 7 in Table A.9)

The IDCOR calculation of TMLB' (no LOCA) case shown in Figure A.7 gives almost the identical pressure in containment, as in Figure A.4 for the TMLB' seal LOCA case. In comparison, the SARP results[9] for Zion TMLB1 and TMLB2 shown on Figures A.8 and A.9, indicate two pressure spikes at core uncovery (1.88 hours) and at RPV failure (2.83 hrs) which are similar to the same event in IDCOR calculations (2.2 and 4.0 hrs, respectively). After RPV rupture, the containment pressure rises slowly for the SARP calculation until the end of SARP calculation). It appears that if the SARP calculations proceeded to about 40 hours the containment would rupture as in the IDCOR case (Figure A.7).

The Zion TMLU case (SARP) in Figure A.10 shows one small pressure peak at core uncovery (2 hrs) and a large 149 psi peak at 3.16 hours, when the containment ruptures because of a hydrogen burn.

<u>SLFC, ALFC and 10-AEFC (S_2D) Accidents</u> (Sequences 9 through 11 in Table A.9)

The containment pressure for SLFC accidents from IDCOR calculations (Figure A.11) shows three peaks; one at 1.8 hours (26 psia), the second at core uncovery (7.2 hours, 23 psi) and the third sharp peak at RPV failure (13.9 hrs, 34 psi). Figure A.12 shows that for ALFC accident (IDCOR results) there is only one main pressure peak at 3.5 hours (35 psi), with some very small pressure peaks at 2.3 hours (RPV failure) since the reactor vessel is already depressurized.

After 5 hours (SLFC) or 15 hours (ALFC) the pressure slowly decreases. Finally, the S_2D-ε accident (SARP results), Figure A.13 shows a number of small peaks between 18 and 25 psi pressure which occurs between 0 and 5.0 hours during the accident, with slowly decreasing pressure after 6 hours. All three accidents (SLFC = $S_2H\delta$, ALFC = $AH\delta$, S_2D-ε) lead to relatively small loading of containment, without final containment failure (NCF), both for IDCOR and SARP calculations.

A.3 Comparison of Accident Releases

The fractional releases of core radioactivity inventory into the environment from the previously discussed accidents are given in Table A.10. From the table, one can see that IDCOR TMLB'-δ-seal LOCA and TMLB'-δ-no LOCA releases are identical, if we compare those release with SARP-seal LOCA cases (see first three lines of Table A.10), we can see that IDCOR releases are one and two orders of magnitude smaller than S_2D C_{ir} and S_2D C_{ir} F_{ir} 1 releases, respectively. On the other hand, the IDCOR TMLB'-δ-seal LOCA releases for iodine (I) and cesium (Cs) are all about 100 to 1,000 times larger than S_2D C_{ir} releases (released at 24 hours) even if they are released later (32 hours). The SARRP and IDCOR predictions for the other elements are similar for the same compared cases (see tellurium (Te), strontium (Sr), ruthenium (Ru), lanthanum (La) columns in Table A.10). Thus, the IDCOR and SARP releases often show remarkable differences, because of different modeling approaches (particularly for I and Cs).

The SARRP uncertainty analysis was performed at BNL for the Zion plant.[11] In the course of calculations, the SURRY BIN matrix was used, depicting the containment performance of the plant. This matrix was extended from 15 bins (originally used for Surry in Ref. 17, Table D2) to 19 bins for Zion, where the last bins 16 to 19 represent the Surry bins 1 to 4 with containment failure because of direct heating of the containment. The bin table is given in Table A.11 and is explained in detail in Ref. 17 for each bin. The code SOURCE was used at BNL, to produce approximate source terms for the 19 Zion bins. The results are given in Table A.12. These releases were obtained from the basic four Source Term Code Package (STCP) releases given in the BCL[10] results of Table A.10. The combinations of the releases for 19 bins were performed by the SOURCE code, which allows calculation of radioactivity releases for early or late containment failure, with sprays operating or failed and with different containment failure modes given in Table A.11.

Several mechanisms have been identified by which there could be significant releases of Cs and I from the containment atmosphere at late times. These include; (a) revolatilization from the RCS, (b) slow deposition of initial releases from the RCS, (c) radioactive decay chains, (d) resuspension because of depressurization at containment failure, (e) retention in the melt until after vessel failure. All of these mechanisms are lumped together as a "delayed release" for I and Cs in SOURCE.

Another important phenomenon not modeled in the STCP methodology deals with radiological releases associated with high-pressure melt ejection and direct containment heating.

If the primary system is pressurized at the time of vessel breach, fuel will be ejected under pressure in a process which can result in significant

aerosol generation, as has been demonstrated experimentally. The SOURCE code also accounts for releases because of early containment failure by direct heating which was not modeled in the BCL results.[10]

In general, much higher releases are predicted by the SOURCE code, than in the STCP calculations. Those higher releases are because of more volatile iodine (I_2, hydrogen iodide, etc.), delayed release of I and Cs from the RCS, and the direct heating release attributed to high-pressure ejection of corium from the RCS following high-pressure sequences.

A.3.1 IDCOR and SARP Modeling Differences

There are a number of differences between the IDCOR and the BCL computer models. These differences are listed in the form of issues, which have been discussed by NRC and IDCOR staff in a series of meetings. These issues are listed in Table A.13. Out of the 18 issues, a subset of 8 have been identified that are appropriate to the subject of this section. Each issue is briefly discussed in the following sections. Differences between IDCOR and BCL will be identified and their significance indicated.

In-Vessel Hydrogen Generation (NRC/IDCOR Issue 5)

There are significant differences between the IDCOR and BCL predictions of hydrogen (H_2) generation during in-vessel core melting. During the early stages of core heatup and degradation (while the fuel rods are still in place in the core region), both IDCOR and BCL predict similar H_2 generation. However, after the fuel rods and cladding begin to melt and relocate in the bottom of the reactor vessel, the BCL analysis indicates substantially, more H_2 generation than the IDCOR analysis. Often twice as much H_2 is generated in the BCL analysis than in the IDCOR analysis.

Hydrogen is important to containment loading because it is a combustible and noncondensable gas. The larger amount of H_2 generated in-vessel in the BCL and SARRP analysis leads to a higher likelihood of hydrogen burning along with a higher likelihood of early containment failure.

Core Slump, Core Collapse, and Reactor Vessel Failure (NRC/IDCOR Issue 6)

The IDCOR core slump model assumes that after 40% of the core has melted, it will relocate into the bottom of the reactor vessel, which will then rapidly fail because of local penetration failure. Thus, only a fraction of the core will be initially released from the reactor vessel. The remainder of the core melts down over a much larger time period. On the other hand, the BCL core slump model assumes total collapse of the core into the bottom of the reactor vessel after 75% of the core is predicted to melt. Thus, all of the core debris is available to be released when the vessel is predicted to fail in the BCL model. This modeling tends to increase the hydrogen available for detonation/deflagration at vessel break (Issue 5), and the potential for direct heating (Issue 8) as well as promoting a more rapid core-concrete attack (Issue 10) with more aerosol generation (Issue 9).

Direct Heating of Containment (Issue 8)

The pressure rise in containment because of direct heating is proportional to the quantity of core debris dispersed from the reactor vessel. The BCL analysis predicts significantly more debris release at vessel failure than the IDCOR analysis. Thus the potential for early containment failure because of direct heating is higher in the BCL analysis.

The assumption that all the core debris is released at vessel failure (BCL analysis) is clearly conservative and the SARRP[11] uncertainty study has addressed the probability of small debris fractions. The IDCOR results appear to be too optimistic considering the lack of supporting large scale experiments. However, spray operation may reduce the pressure pulse associated with direct heating by flooding the reactor cavity which may help quench the core debris. Containment sprays and containment cooling are thus very important to the timing and mode of containment failure.

Ex-Vessel Heat Transfer Model from Molten Core to Concrete (Issue 10)

Almost all cases analyzed by IDCOR assume that the reactor cavity is flooded by water from containment sprays. This mechanism allows hot debris to drop from the failed RPV into water and form a coolable debris bed under water. The solidified debris is assumed to stay solid without further attacking the concrete. In the BCL analyses the spray is assumed to fail and the debris attacks concrete continually.

The IDCOR assumption appears to be overly optimistic given the lack of large scale debris coolability data. If the debris is not cooled, it will attack concrete and produce noncondensable gases, even if the top surface of the debris is covered by water. In view of the uncertainty, the slow pressurization of the containment resulting from the coolable debris assumption in the IDCOR analyses appears to be unjustified and the potential for the rapid containment pressurization predicted in the BCL analyses should be considered. The SARRP uncertainty study[11] has assigned weighting factors to the various possible combinations of containment spray operation and debris cooling. Most of the risk was found to come from early containment failure because of direct heating.

Containment Failure Because of Hydrogen Deflagration (Issue 17)

In the IDCOR analysis hydrogen deflagrations do not occur. This is because of the small amounts of hydrogen generated (about 1/2 of the BCL[9,10] predictions).

BCL predicts substantial hydrogen generation and subsequent combustion but even with hydrogen combustion the containment pressure is substantially below the failure point.

Containment Performance (Issue 15)

For those cases with failure of the containment heat removal system, the containment eventually overpressurizes and IDCOR assumes that a relatively small containment failure will occur which allows gradual leakage of the containment atmosphere to the auxiliary building. By comparison the BCL analysis

assumes an opening large enough (7 ft^2) to rapidly depressurize the primary containment. These differences result in substantial differences in the estimated decontamination factor for the auxiliary building.

Fission-Product Release Prior to Vessel Failure (Issue 1)

Both studies predict similar releases of the more volatile fission products during in-vessel core degradation with the exception of Te. (IDCOR predicts about 1/10 the TE releases of BCL for TMLB'-δ). However, in the IDCOR analysis, rubidium (Rb), zirconium (Zr), plutonium (Pu), lanthanum (La), barium (Ba), yttrium (Y), technetium (Tc), rhodium (Rh), and palladium (Pd) were omitted from the in-vessel release. This IDCOR omission will lead to somewhat smaller environmental doses than in the BCL cases. IDCOR has agreed to account for these omissions in future calculations.

Fission-Product and Aerosol Retention in the Primary System (Issue 4)

Differences in the primary system retention predicted by IDCOR and BCL are not too significant and differ by less than a factor of two. In the BCL modeling it is assumed that fission products retained in the primary system at the point of vessel failure are retained permanently. In the IDCOR analysis of the Peach Bottom plant, revaporization of these fission products after vessel failure was modeled.[18] This revaporization is not used in the IDCOR model for Zion since the higher heat losses from the primary system are predicted to keep the Zion primary system relatively cool.

Ex-Vessel Fission-Product Release (Issue 9)

There are significant differences between the IDCOR and BCL analyses for fission product release as a result of core-concrete interactions. The higher releases of Sr (and also La, Ce and Ba groups) in Table A.10 in the SARP analyses are because of the modeling of ex-vessel fission-product release. Fission-product release and inert aerosol generation during core-concrete interactions was not modeled in the IDCOR analysis. IDCOR argued that even for a dry cavity the aerosol generation during core-concrete interactions would increase aerosol density in containment and increase aerosol agglomeration or settling, thus reducing the predicted environmental release fractions relative to those predicted without this additional aerosol source. BNL does not consider that this IDCOR argument has been adequately supported. In addition, the IDCOR predicted core debris temperatures during core-concrete interactions are very high. Based on experimental evidence one would expect the release of some of the refractory fission-product groups at these elevated temperatures. The BCL and SARRP analyses currently model the release of the refractory fission products and the inert aerosols and the environmental release fractions are significant (refer to Table A.10).

A.4 Offsite Consequences

The estimated risk results for Zion are summarized in Table A.15. The IDCOR results indicate that the large-volume containment is very effective in mitigating fission-product releases and the offsite consequences are quite small. However, the SARRP[11] consequence results are very different than IDCOR for Zion.

The major differences between SARRP and IDCOR appear to be containment performance and releases during core-concrete interaction. The overall containment failure probability produced by SARRP is shown in Figure A.12. SARRP[11] estimates that there is about a 4% chance (mean) of early containment failure for Zion core-damage sequences. A comparison of containment failure probabilities is given in Table A.15. As shown in Figures A.14 and A.15, SARRP[11] predicts that there is only about a 2% chance of late containment failure as long as one or more containment heat removal systems (F and C for fans and sprays, respectively) are operational. For the SE damage state there is no water in reactor cavity so there is a substantial likelihood of containment failure (SARRP estimates a range of 5 to 80%). For late containment failure sequences the releases from core-concrete interactions tend to be important contributors to risk. IDCOR assessed the probability of early containment failure to be negligible. The overall SARRP risk profile appears to be fairly consistent with previous studies of Zion. However, IDCOR is two orders of magnitude lower.

A.5 Summary and Risk Insights

A.5.1 Core-Damage Profile

Generally, as in other PWRs, transients (including station blackout), service water failure, small LOCA and ATWS (including SGTR and ISL) dominate the core-damage risk profile for the Zion plant. It is particularly noteworthy that loss of component cooling water which may lead to a RCP seal LOCA sequence under high-pressure injection failure is the most predominant contributor to the core damage in both the Zion review and the SNL study. This is a result of two conservative considerations in (a) the quantification of the CCWS pipe rupture frequency and (b) the assumption that loss of component cooling water leads very promptly to a sizable seal LOCA. It is also noted that the small LOCA is an important contributor to core damage. It is also important to note that the major differences in quantitative results among the studies result from the difference in modeling assumptions and level of detail and scope rather than plant differences or data differences.

Although there are a large number of contributing sequences there is a reasonable expectation that if these sequence types can be minimized, then the overall core damage frequency will be controlled. As in References 8 and 19, this principle is used in Sections 3 and 4 of this report to develop guidelines and criteria to control the overall CDF (Goal 3).

It is recognized that the qualitative accident sequence descriptors are rather general and broad and that similar hardware functional requirements and operator actions in the large-volume plants would lead to the same general accident sequences. A plant-specific examination (such as a failure mode and effects analysis coupled with a fault tree/event tree analysis) is needed in order to identify the plant specific vulnerabilities (e.g., in maintenance practices, operator training, and emergency operating procedures) to contribute to a given general sequence descriptor.

A.5.2 Consequence Analysis

The assessment of core meltdown phenomena and containment response indicates that the large dry containment provides a robust defense against early

severe accident pressures and temperatures. However, differences in the IDCOR and BCL/SARRP assessments of containment response and fission product release result in major differences in the predicted offsite consequences. The IDCOR assessment indicates that the containment will only fail very late in the accident sequence if it fails at all. BCL/SARRP predict the possibility of much earlier containment failure and much higher releases. The overall risk estimated[11] for Zion is dominated by the early containment failure sequences which in turn are dominated by the perceived possibility of direct containment heating.

If the containment does not fail early, SARRP considers the likelihood of containment failure to be low (about 5%) because of the redundant and diverse containment heat removal systems and the availability of long-term coolant supplies.

A.6 References

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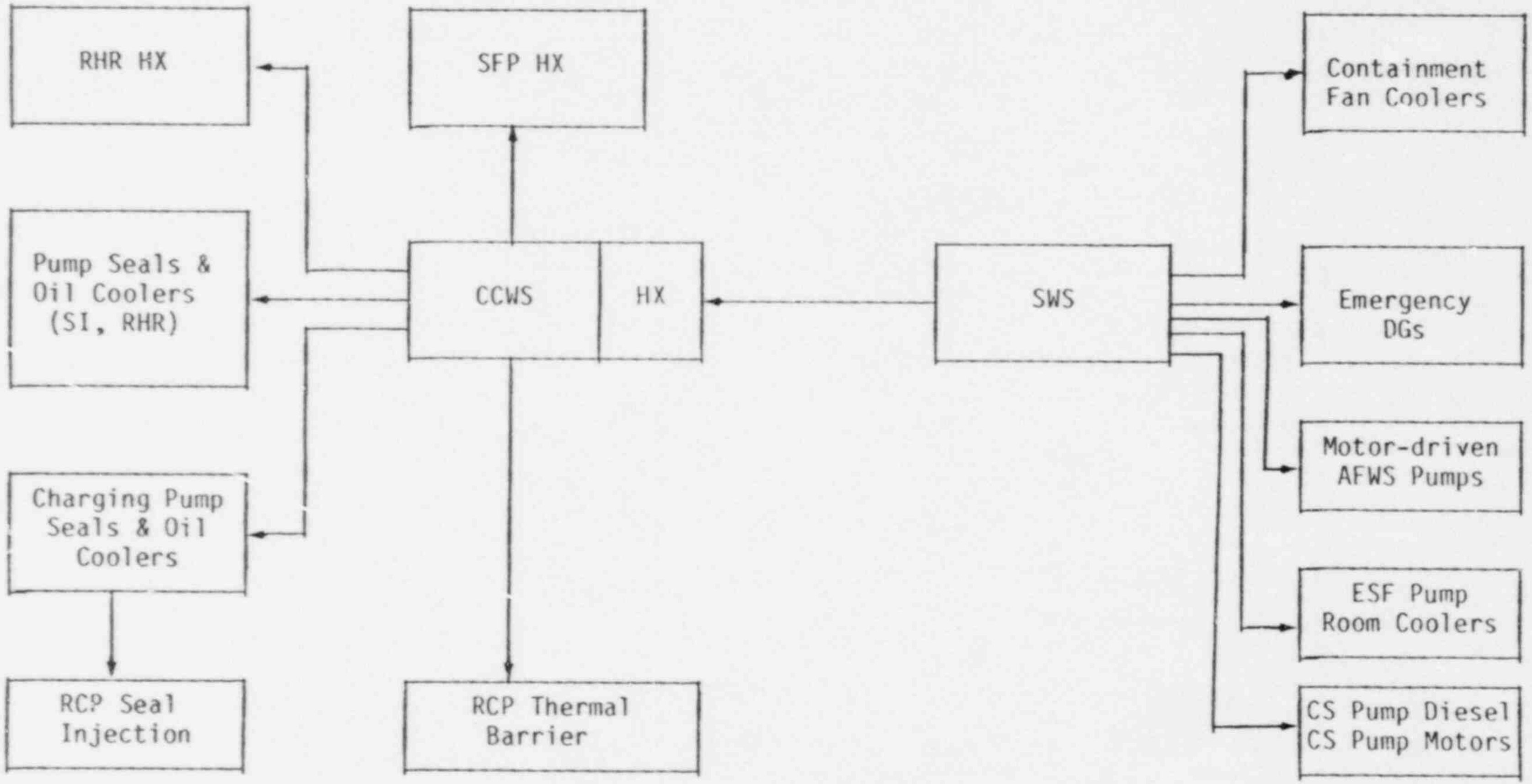

Figure A.1 A heat and fluid flow schematic for Zion.

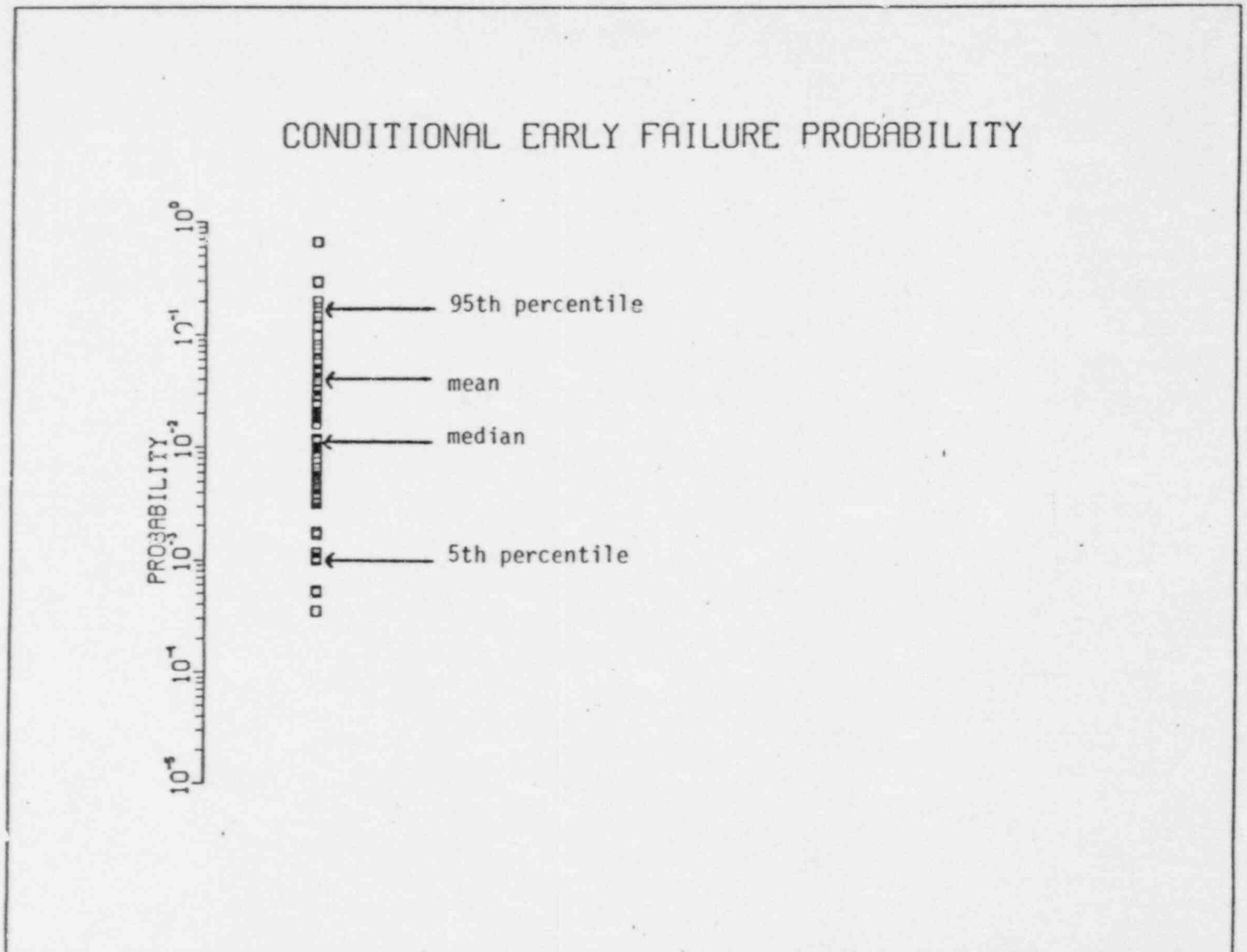

Figure A.2 Conditional probability of early containment failure: all sequences included.[11]

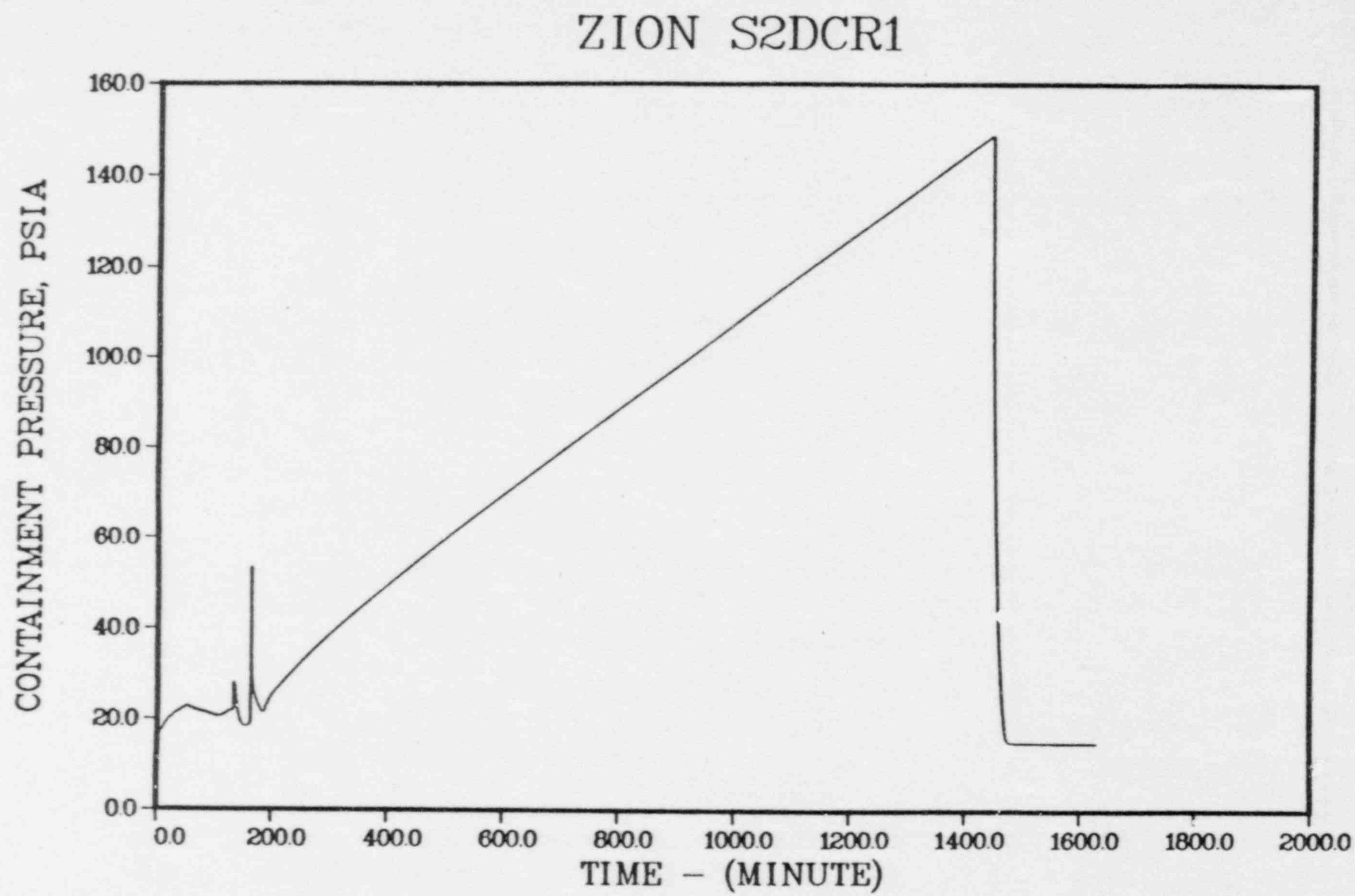

Figure A.3 Containment pressure response for S_2DC_R sequence (SARP).[10]

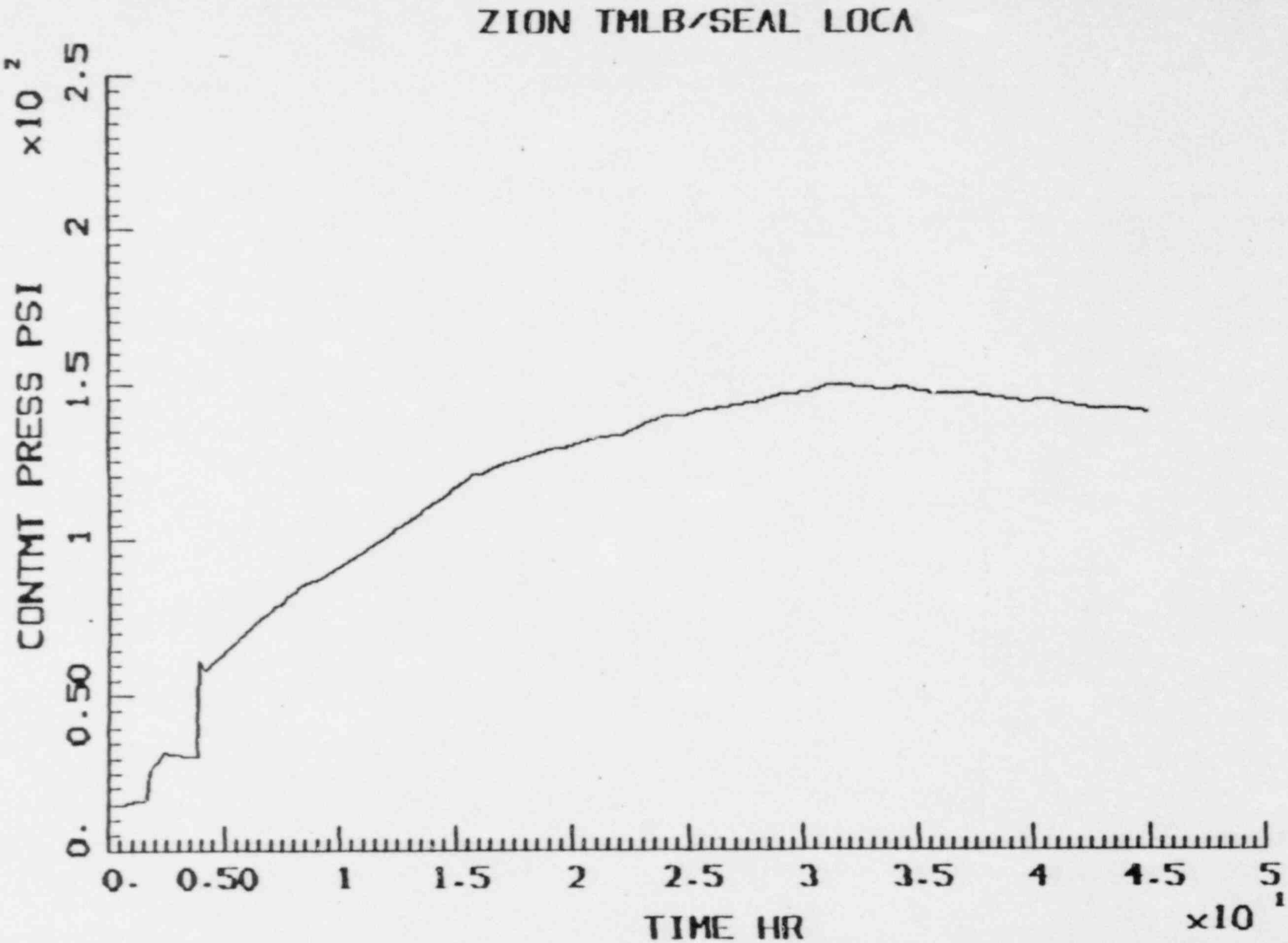

Figure A.4 Containment building pressure (IDCOR).[14] (Note the scale $2.5x10^2$ psi and $5x10^1$ hours, used in all IDCOR figures.)

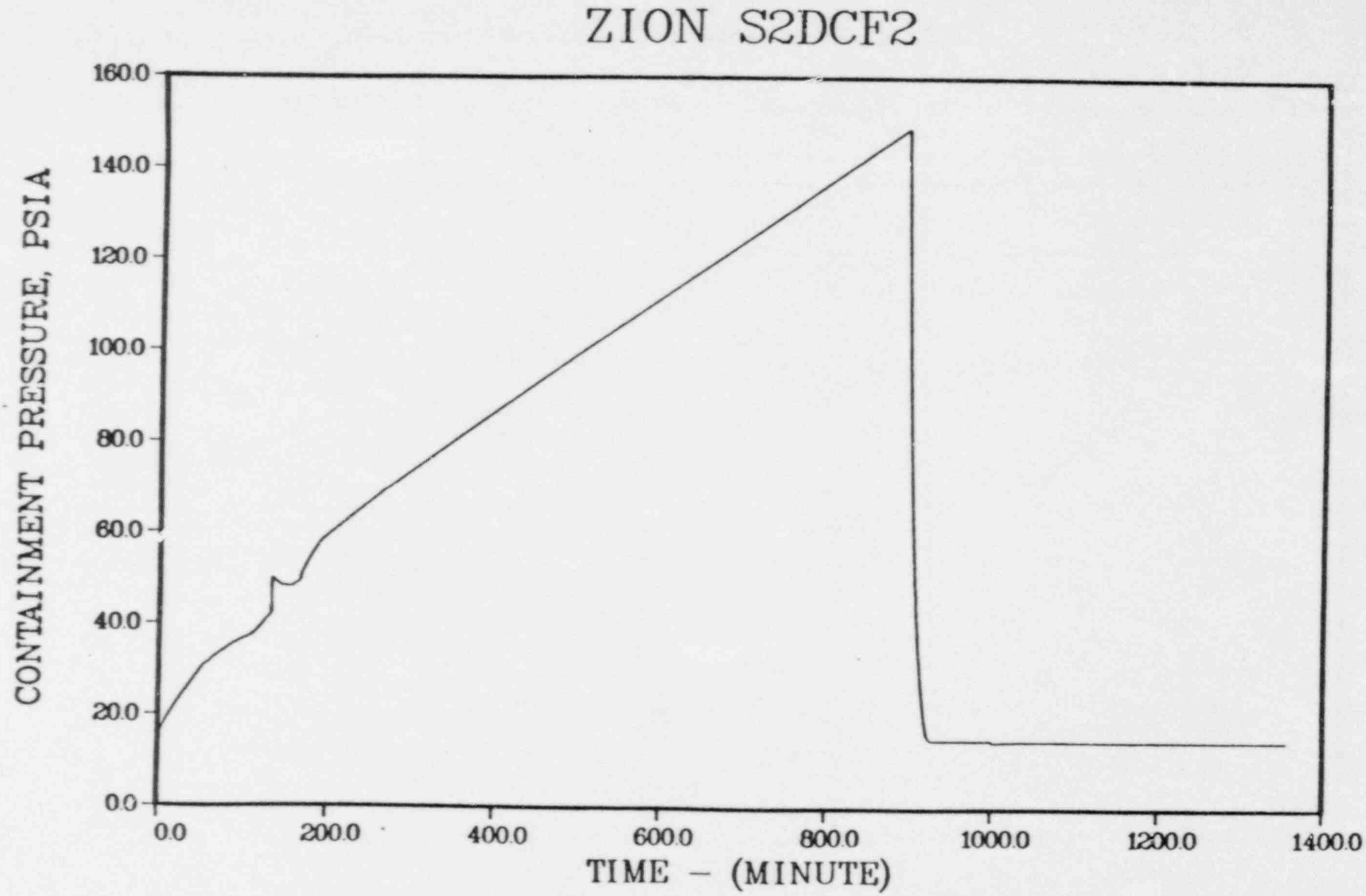

Figure A.5 Containment pressure response for S_2DC_R sequence with late containment failure (SARP).[10]

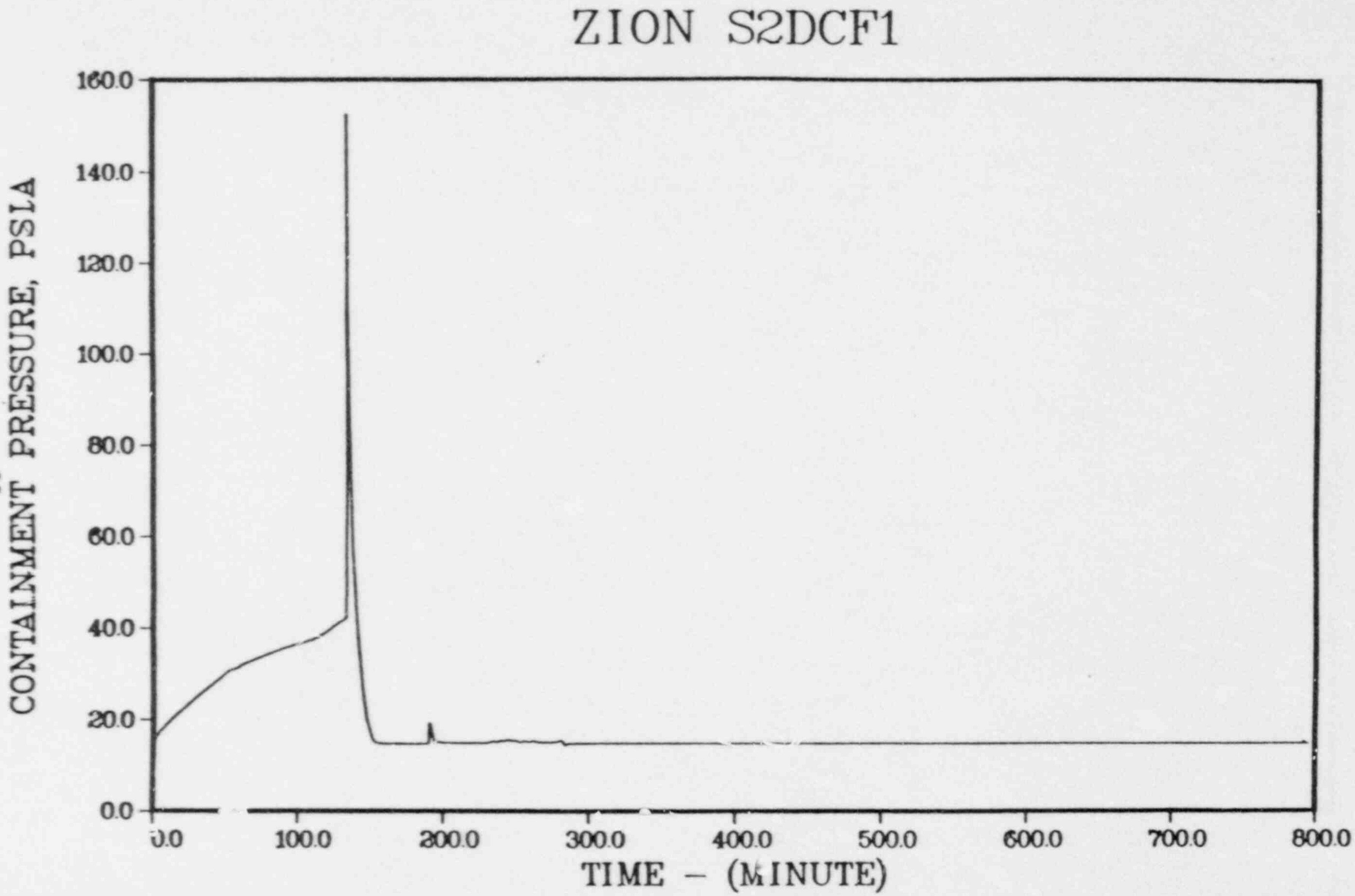

Figure A.6 Containment pressure response for S_2DC_R sequence with early containment failure (SARP).[10]

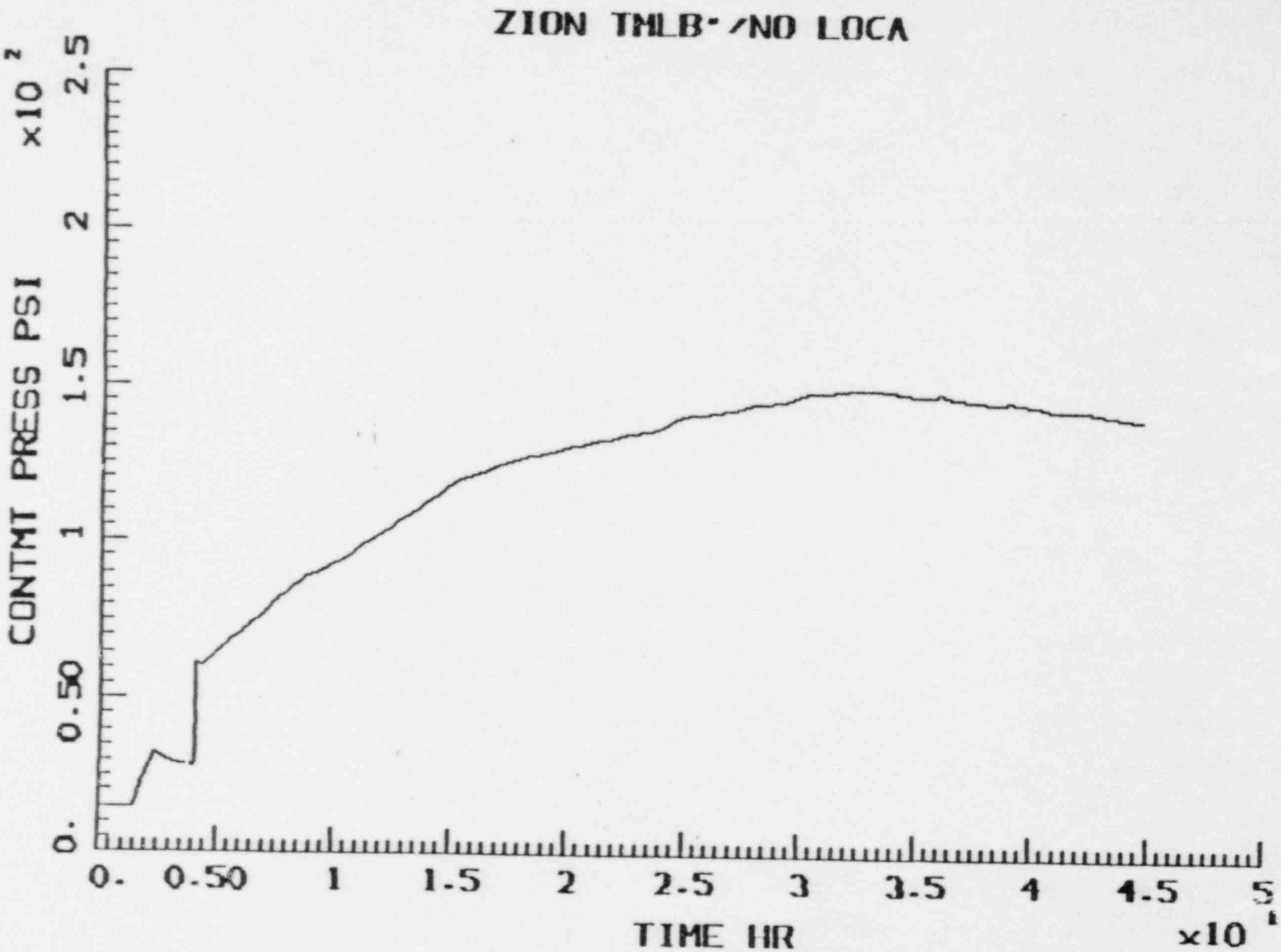

Figure A.7 Containment building pressure (IDCOR).[14]

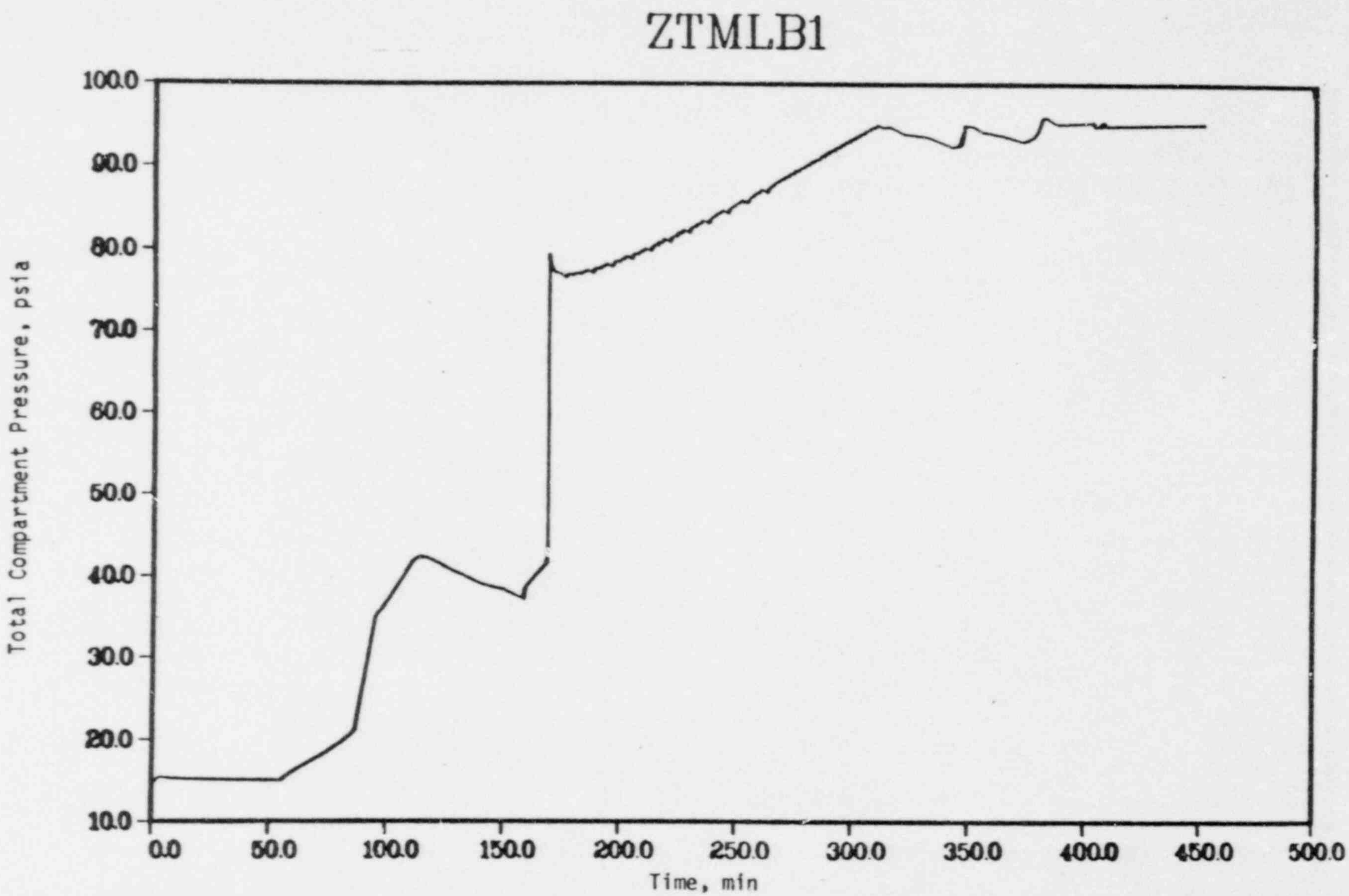

Figure A.8 Containment pressure response for Zion TMLB' sequence with rapid debris quench (SARP).[9]

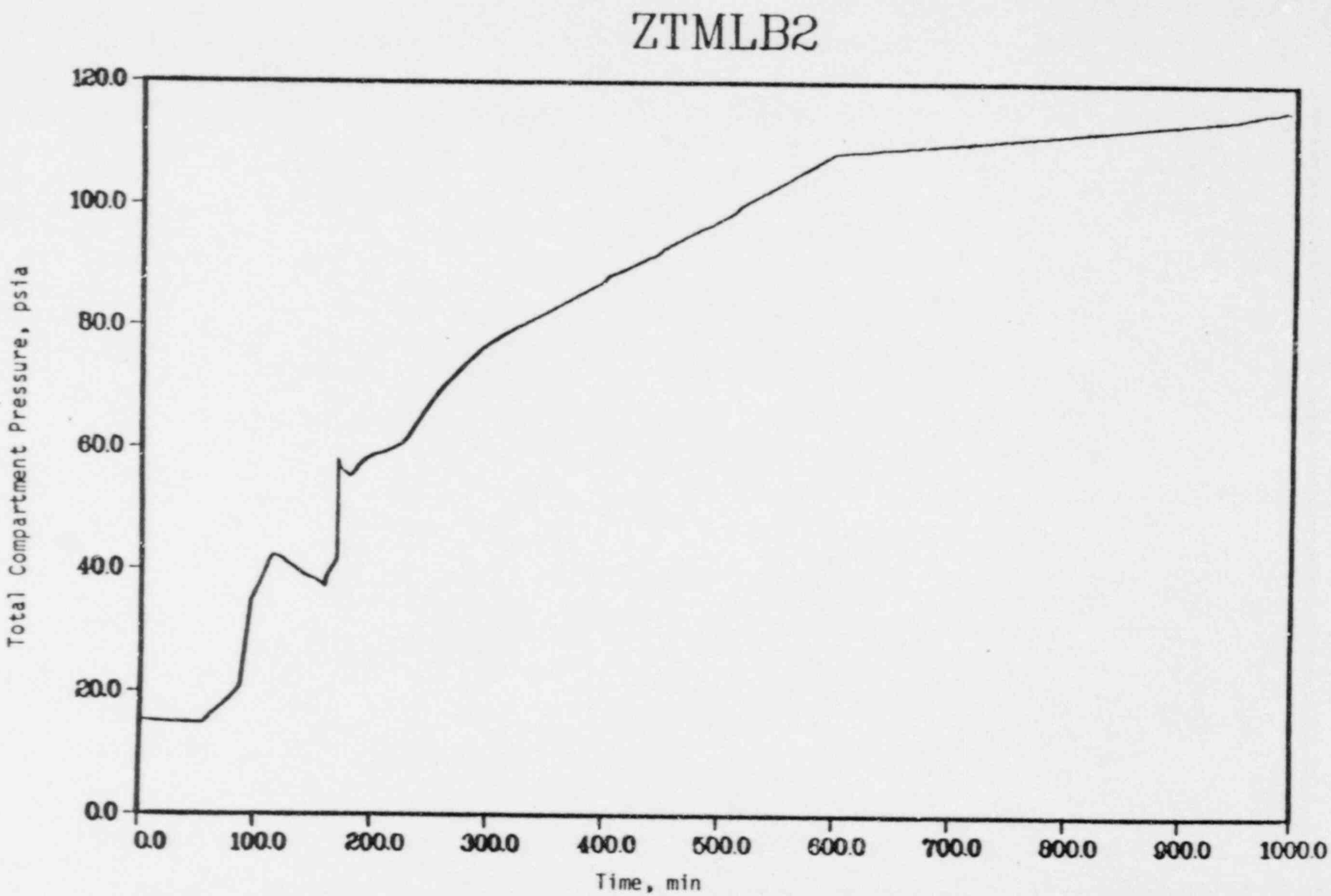

Figure A.9 Pressure in containment for Zion TMLB' sequence with direct concrete attack (SARP).[9]

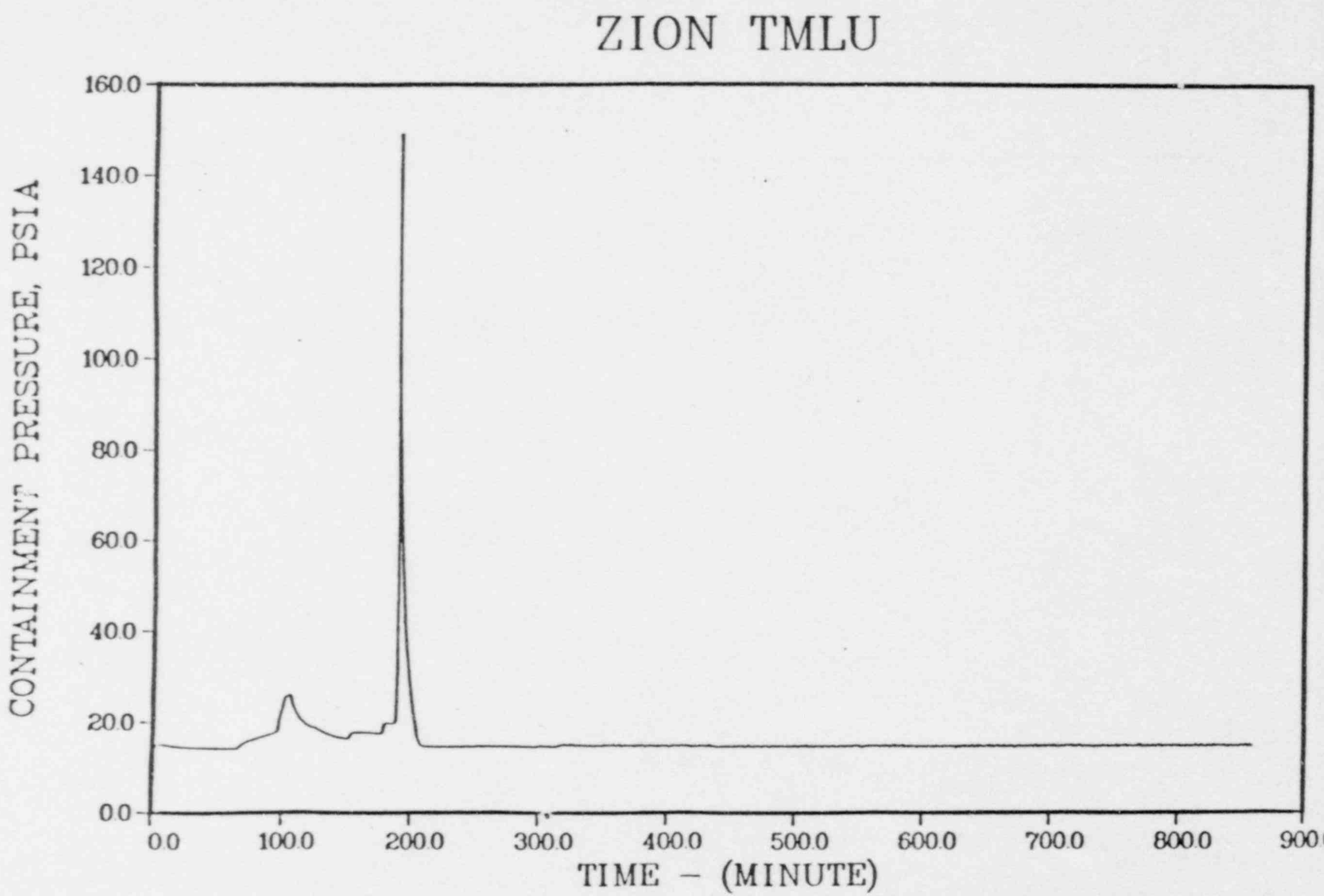

Figure A.10 Containment pressure response for the TMLU sequence (SARP).[10]

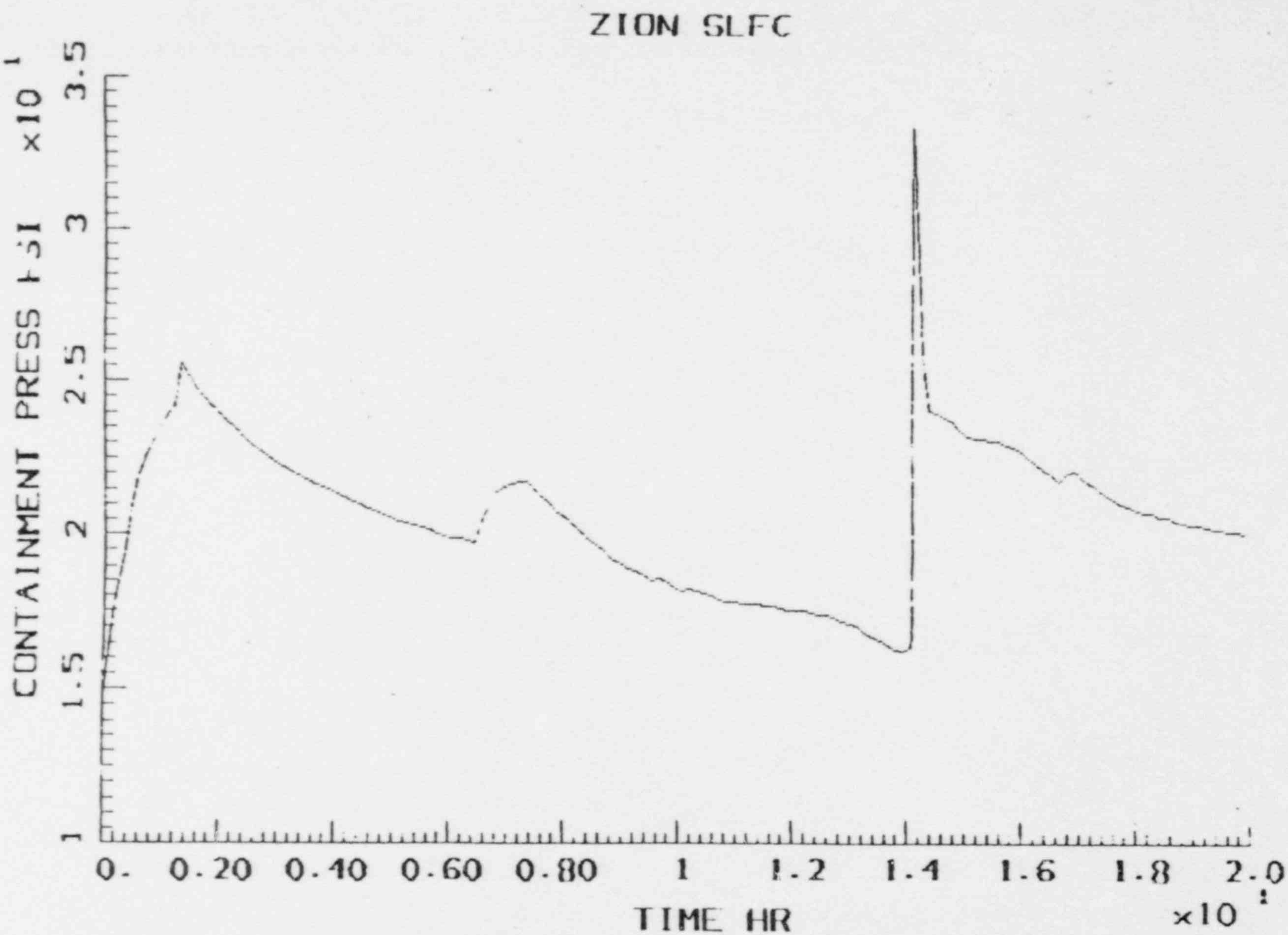

Figure A.11 SLFC containment pressure (IDCOR).[14]

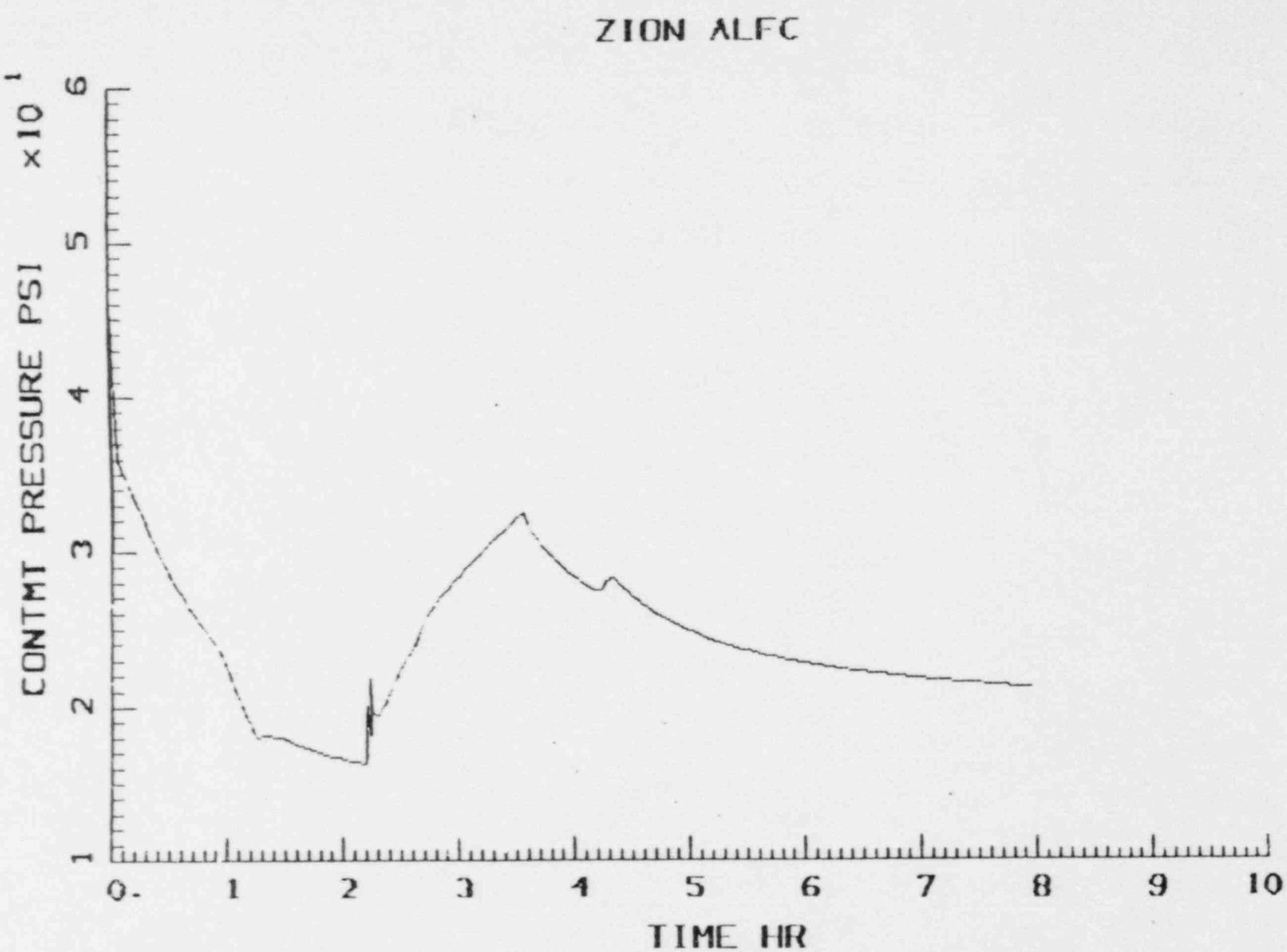

Figure A.12 ALFC containment pressure (IDCOR).[14]

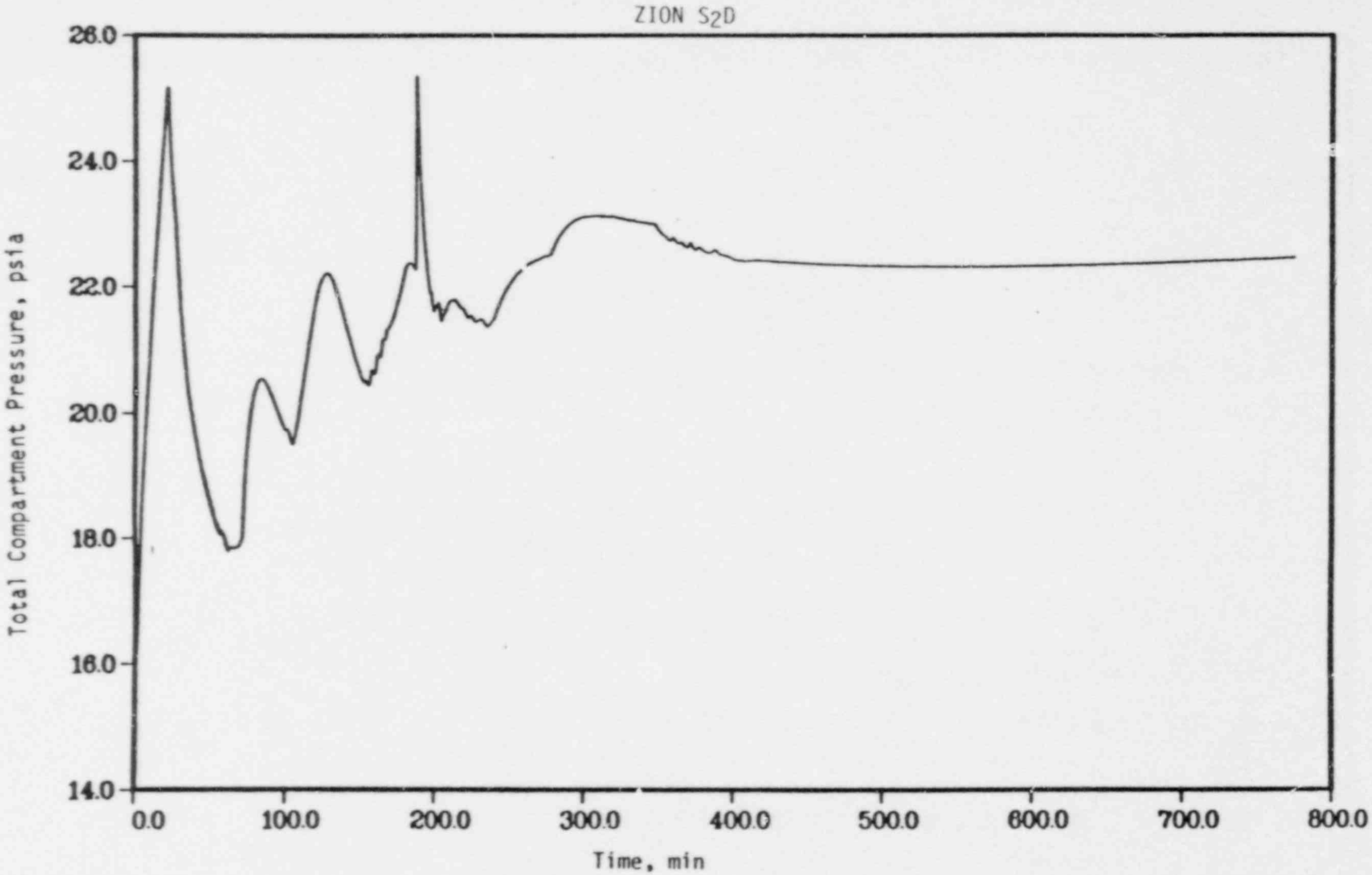

Figure A.13 Containment pressure response for Zion S_2D sequence with containment sprays and concrete attack by core debris (SARP).[9]

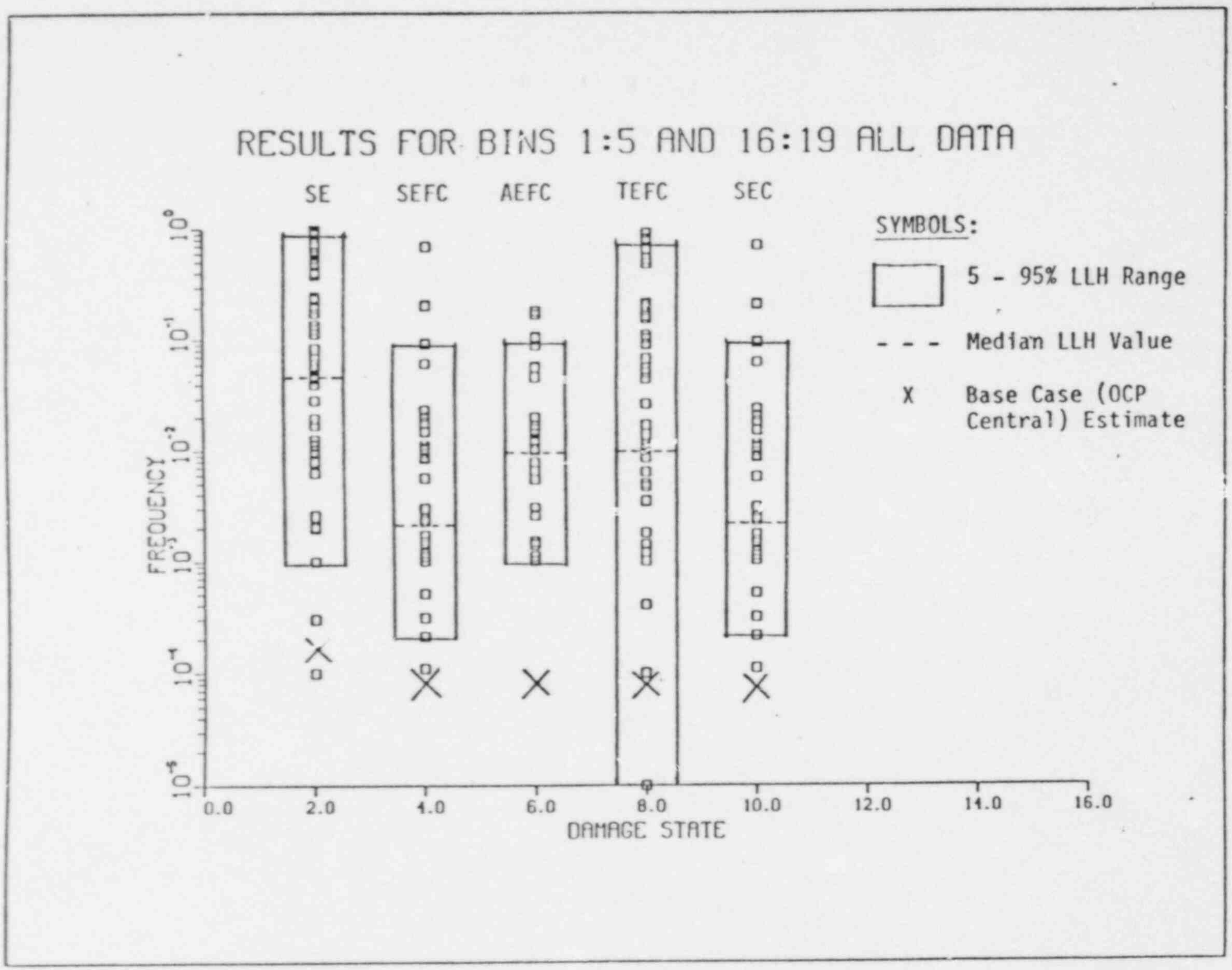

Figure A.14 Comparison of LLH and point-estimate results for conditional probability of early containment failure (Bins 1 - 5 and 16 - 19) for five representative plant damage states.[11]

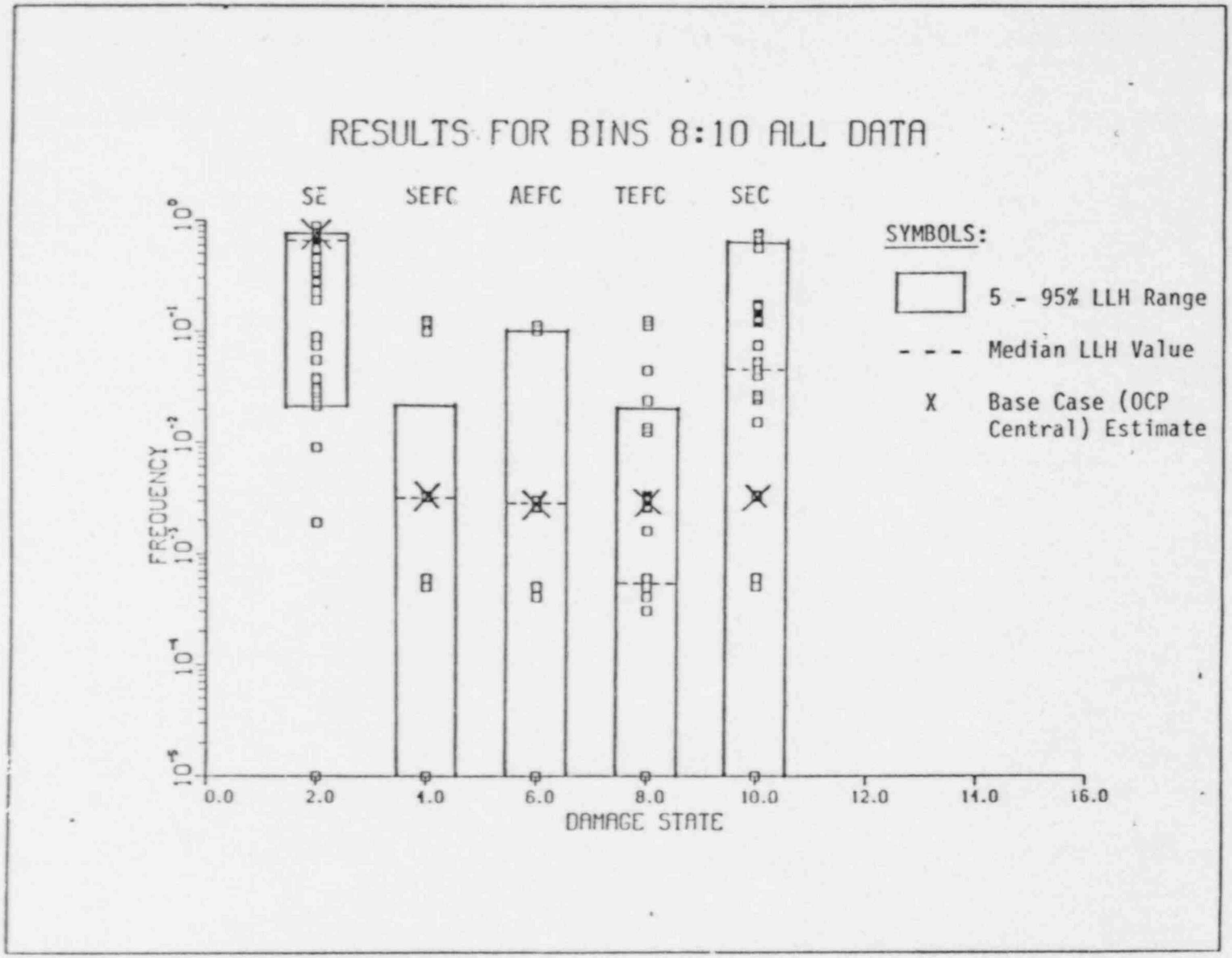

Figure A.15 Comparison of LLH and point-estimate results for conditional probability of late containment failure (Bins 8 - 10) for five representative plant damage states.[11]

Table A.1 Design Comparison of Nuclear Power Plants With Large-Volume Ccontainments[1]

	Zion	Surry	Oconee
Owner	Commonwealth Edison	Virginia Electric & Power	Duke Power
Site	Lake County, Illinois	Surry County, Virginia	Oconee County, South Carolina
Capacity	3250 MWt	2441 MWt	2568 MWt
Type	4 Loop Westinghouse PWR (2 Units)	3 Loop Westinghouse PWR (2 Units)	2 Loop Babcock & Wilcox PWR (3 Units)
Containment	Large, Dry	Large, Dry, Subatmospheric	Large, Dry
LPIS & LPRS	4 Accumulators 2 RHRS Pumps	3 Accumulators 2 RHRS Pumps[2]	2 Core Flood Tanks 3 DHRS Pumps
CSIS & CSRS	3 CS Pumps[3]	2 CS Pumps	2 RBS Pumps
HPIS & HPRS	2 SI Pumps 2 Centrifugal Charging Pumps	3 Charging Pumps	3 Centrifugal Charging Pumps
CHRS	5 RCFC Fan/Cooler Units	4CSRS Pumps with 4 CHRS HXs	3 RBCS Fan/Cooler Units
AFWS	2 Motor-Driven Pumps 1 Turbine-Driven Pump	2 Motor-Driven Pumps 1 Turbine-Driven Pump	2 Motor-Driven Pumps 1 Turbine-Driven Pump
EPS	3 Essential Power Divisions with 2 (Dedicated) + 1 (Swing) Diesel Generators	2 Essential Power Divisions with 1 (Dedicated) + 1 (Swing) Diesel Generators	3 Load Divisions with 2 Hydro-electric Generators and 1 Transmission System from a Steam Station
CCWS	5 Pumps & 3 HXs Shared by Both Units	4 Pumps & 4 HXs Shared by Both Units	6 Low Pressure SW Pumps Shared by 3 Units with a Backup from
ESWS	6 Pumps Shared by Both Units	8 Pumps & 3 Diesel Pumps Shared by Both Units	3 High Pressure SW Pumps[4]

[1]Design parameters are for one unit unless otherwise noted.
[2]No associated heat exchangers in recirculation loops.
[3]The 3rd pump is a direct-driven diesel pump.
[4]Component Cooling Water System in Oconee is used mainly for cooling of the RCP thermal barrier.

Table A.2 IDCOR Baseline Core-Damage Profile

Rank	Sequence	Core-Damage Frequency (yr^{-1})	Plant-Damage State
1	Small LOCA, failure of recirculation cooling	1.6E-5(1)	SLFC
2	Seismic, loss of all ac power	5.6E-6	SE
3	Large LOCA, failure of recirculation cooling	4.9E-6	ALFC
4	Medium LOCA, failure of recirculation cooling	4.9E-6	ALFC
5	Fire in the AEE room with safety system malfunctions	2.8E-6	TEFC
6	Fire in the cable spreading room with loss of fans	1.8E-6	TEC
7	Spurious safety injection, failure to control safety injection, recirculation cooling	1.5E-6 (1.64E-6)(2)	SLFC
8	Spurious safety injection, loss-of-offsite power, loss of ESF Buses 148 and 149	1.5E-6 (1.43E-6)(2)	SLC
9	Large LOCA, failure of low-pressure injection	1.3E-6	AEFC
10	Medium LOCA, failure of low-pressure injection	4.3E-7 (4.36E-7)(2)	AEFC
11	Loss of main feedwater, loss of offsite power, loss of ESF Buses 148 and 149	2.9E-7	TEC
12	Reactor trip, loss-of-offsite power, loss of ESF Buses 148 and 149, auxiliary feedwater failure	2.1E-7 (2.23E-7)(2)	TEC
13	Turbine trip, loss-of-offsite power, loss of ESF Buses 148 and 149, auxiliary feedwater failure	2.1E-7	TEC
14	Turbine trip due to loss-of-offsite power, loss of all ac power, auxiliary feedwater failure	2.0E-7	TE(TMLB')

Table A.2 (Continued)

Rank	Sequence	Core-Damage Frequency (yr^{-1})	Plant Damage State
15	Loss of main feedwater, auxiliary feedwater failure, failure of feed and bleed cooling	1.4E-7 (1.33E-7)(2)	TEFC
16	Interfacing systems LOCA (Residual Heat Removal System Inlet Valves)	1.1E-7	VL
17	Medium LOCA, failure of low pressure injection, containment spray injection and containment fan coolers	6.9E-12	AE
18	Spurious safety injection, failure of recirculation cooling, failure of containment sprays	2.5E-9	SLF
Totals		4.2E-5 (3.2E-5)(3)	

(1) 1.6E-5 = 1.6×10^{-5}.
(2) Values shown inside parentheses correspond to those in ZPSS.
(3) Internal events only.

Table A.3 Zion Review Core-Damage Profile

Rank	Sequence	Core-Melt Frequency (yr^{-1})	Plant-Damage State
1	CCW failure, causing failure of all charging and SI pumps, seal LOCA	2.0E-4	SEFC
2	Loss of offsite power, failure of component cooling water, failure to recover offsite power in four hours (recovery prior to eight hours)	4.6E-5	SEFC
3	Loss of offsite power, failure of component cooling water, failure to recover offsite power in one hour (recovery prior to four hours)	4.0E-5	SEFC
4	Loss of offsite power, failure of component cooling water, failure to recover offsite power in eight hours, failure of containment fans	1.8E-5	SEC
5	Small LOCA, failure of recirculation cooling	1.6E-5	SLF
6	Loss of offsite power, failure of component cooling water, failure to recover offsite power in eight hours	7.9E-6	SEFC
7	Failure of dc bus 112, causing failure of one PORV and loss of ac bus 149, failure of auxiliary feedwater	7.0E-6	TEFC
8	Seismic, loss of all ac power	5.6E-6	SE
9	Large LOCA, failure of recirculation cooling	4.9E-6	ALF
10	Medium LOCA, failure of recirculation cooling	4.9E-6	ALF
11	Loss of offsite power, failure of component cooling water, failure to recover offsite power in eight hours, failure of containment sprays and fan coolers	4.7E-6	SE
12	Large LOCA, failure of low pressure injection	1.4E-6	AEFC

Table A.3 (Continued)

Rank	Sequence	Core-Melt Frequency (yr^{-1})	Plant-Damage State
13	Loss of offsite power, failure of auxiliary feedwater, failure of feed and bleed, failure to restore offsite power in four hours (recovery prior to eight hours)	1.1E-6	TEFC
14	Loss of offsite power, failure of auxiliary feedwater, failure of feed and bleed, failure to restore ac power in one hour (recovery prior to four hours)	1.0E-6	TEFC
15	Interfacing system LOCA	1.1E-7	V
Total		3.6E-4 (3.54E-4)*	

*Internal events only.

Table A.4 Comparisons of Dominant Accident Sequences Obtained by SARP Rebaseline, Zion Review and IDCOR-Baseline

Rank					
SARP Rebase-line	Zion Review	IDCOR Baseline	Sequence	Core-Melt Frequency (yr^{-1})	Plant-Damage State
1	1	-	CCW failure, causing failure of all charging and SI pumps, seal LOCA	1.2E-4 (2.0E-4)*	SEFC (SEFC)*
2	5	1	Small LOCA, failure of recirculation cooling	1.6E-5 (1.6E-5) ((1.6E-5))*	SLFC (SLF) ((SLFC))*
-	-	2	Seismic, loss of all ac power	((5.6E-6))	((SE))
3	9	3	Large LOCA, failure of recirculation cooling	4.9E-6 (4.9E-6) ((4.9E-6))	ALFC (ALF) ((ALFC))
4	10	4	Medium LOCA, failure of recirculation cooling	4.9E-6 (4.9E-6) ((4.9E-6))	ALFC (ALF) ((ALFC))
-	-	5	Fire in the AEE room with safety system malfunctions	((2.8E-6))	((TEFC))
5	14	-	Loss of offsite power, failure of AFWS, failure of feed and bleed, failure to restore ac power in one hour (recovery by four hours)	2.1E-6 (1.0E-6)	TEFC (TEFC)
-	-	6	Fire in the cable spreading room with loss of fans	((1.8E-6))	((TEC))
-	-	7	Spurious safety injection, failure to control safety injection, recirculation cooling	((1.5E-6))	((SLFC))
-	-	8	Spurious safety injection, loss of offsite power, loss of ESF buses 148 and 149	((1.5E-6))	((SLC))
6	12	9	Large LOCA, failure of low pressure injection	1.4E-6 (1.4E-6) ((1.3E-6))	AEFC (AEFC) ((AEFC))
7	13	-	Loss of offsite power, failure of AFWS, failure of feed and bleed, failure to restore ac power in four hours (recovery by eight hours)	4.6E-7 (1.1E-6)	TEFC (TEFC)

Table A.4 (Continued)

Rank: SARP Rebase-line	Zion Review	IDCOR Baseline	Sequence	Core-Melt Frequency (yr^{-1})	Plant-Damage State
-	-	10	Medium LOCA, failure of low pressure injection	((4.3E-7))	((AEFC))
8	3	-	Loss of offsite power, CCW/SWS loss, failure to restore ac in one hour (recovery by four hours)	3.2E-7 (4.0E-5)	SEFC (SEFC)
9	-	-	Same as sequence 8, only this represents the SWS common mode portion of the rebaselined Zion review sequence no. 3	3.0E-7 (----)	SE (--)
-	-	11	Loss of main feedwater, loss of offsite power, loss of ESF buses 148 and 149	((2.9E-7))	((TEC))
-	-	12	Reactor trip, loss of offsite power, loss of ESF Buses 148 and 149, AFW failure	((2.1E-7))	((TEC))
-	-	13	Turbine trip, loss of offsite power, loss of ESF Buses 148 and 149, AFW failure	((2.1E-7))	((TEC))
-	-	14	Turbine trip due to loss of offsite power, loss of all ac power, AFW failure	((2.0E-7))	((TE or TMLB'))
10	11	-	Loss of offsite power, CCW/SWS loss, failure to restore ac power in eight hours, failure of containment sprays and fan coolers	2.0E-7 (4.7E-6)	SE (SE)
11	2	-	Loss of offsite power, CCW/SWS loss, failure to restore ac power in four hours (recovery by eight hours)	1.5E-7 (4.6E-5)	SEFC (SEFC)
12	-	-	Loss of offsite power, failure of SWS, failure to restore ac power in eight hours. This sequence represents the SWS portions of the rebaselined Zion review sequences no. 4 and no. 6	1.5E-7 (----)	SE (--)

Table A.4 (Continued)

Rank					
SARP Rebase-line	Zion Review	IDCOR Baseline	Sequence	Core-Melt Frequency (yr^{-1})	Plant-Damage State
-	-	15	Loss of main feedwater, AFW failure, failure of feed and bleed cooling	((1.4E-7))	((TEFC))
13	4	-	Same as sequence 12 above, only this is the CCW portion of the rebaselined Zion review sequence no. 4	1.0E-7 (1.8E-5)	SEC (SEC)
14	-	16	Interfacing systems LOCA	1.0E-7 (1.0E-7) ((1.1E-7))	V (V) ((VL))
15	7	-	Failure of dc Bus 112, causing loss of one PORV and loss of ac Bus 148, failure of AFW	5.0E-8 (7.0E-6)	TEFC (TEFC)
16	-	-	Same as sequence 11, only this represents the SWS common mode portion of the rebaselined Zion review sequence no. 2	4.8E-8 (----)	SE (--)
17	6	-	Loss of offsite power, CCW failure, failure to recover ac power in eight hours	3.7E-8 (8.0E-6)	SEFC (SEFC)
-	-	17	Medium LOCA, failure of low pressure injection, containment spray injection and containment fan coolers	((6.9E-12))	((AE))
-	-	18	Spurious safety injection, failure of recirculation cooling, failure of containment sprays	((2.5E-9))	((SLF))
Total**				1.51E-4 (3.53E-4) ((3.2E-5))	

*Information shown inside a single parenthesis and double parentheses corresponds respectively to that obtained by the Zion Review and IDCOR Baseline.

**Internal events only.

Table A.5 Comparison of Dominant Core-Damage Frequency for Zion (Per Reactor Year)

No.	Sequence	SARP Rebase-line	IDCOR Baseline	Plant-Damage State
1	CCW failure, causing failure of all charging and SI pumps, seal LOCA	1.2E-4	---*	SEFC
2	Small LOCA, failure of recirculation cooling	1.6E-5	1.6E-5	SLFC
3	Large LOCA, failure of recirculation cooling	4.9E-6	4.9E-6	ALFC
4	Medium LOCA, failure of recirculation cooling	4.9E-6	4.9E-6	ALFC
5	Loss of offsite power, failure of AFWS, failure of feed and bleed, failure to restore ac power in one hour (recovery by four hours)	2.1E-6	---	TEFC
-	Spurious safety injection, failure to control safety injection, recirculation cooling	---	1.5E-6	SLFC
-	Spurious safety injection, loss of offsite power, loss of ESF buses 148 and 149	---	1.5E-6	SLC
6	Large LOCA, failure of low pressure injection	1.4E-6	1.3E-6	AEFC
7	Loss of offsite power, failure of AFWS, failure of feed and bleed, failure to restore ac power in four hours (recovery by eight hours)	4.6E-7	---	TEFC
-	Medium LOCA, failure of low pressure injection	---	4.3E-7	AEFC
8	Loss of offsite power, CCW/SWS loss, failure to restore ac in one hour (recovery by four hours)	3.2E-7	---	SEFC
9	Same as sequence 8, only this represents the SWS common mode portion of the rebaselined Zion Review sequence no. 3	3.0E-7	---	SE
-	Loss of main feedwater, loss of offsite power, loss of ESF buses 148 and 149	---	2.9E-7	TEC
-	Reactor trip, loss of offsite power, loss of ESF Buses 148 and 149, AFW failure	---	2.1E-7	TEC

Table A.5 (Continued)

No.	Sequence	SARP Rebase-line	IDCOR Baseline	Plant-Damage State
-	Turbine trip, loss of offsite power, loss of ESF Buses 148 and 149, AFW failure	---	2.1E-7	TEC
-	Turbine trip due to loss of offsite power, loss of all ac power, AFW failure	---	2.0E-7	TE --
10	Loss of offsite power, CCW/SWS loss, failure to restore ac power in eight hours, failure of containment sprays and fan coolers	2.0E-7	---	SE
11	Loss of offsite power, CCW/SWS loss, failure to restore ac power in four hours (recovery by eight hours)	1.5E-7	---	SEFC
12	Loss of offsite power, failure of SWS, failure to restore ac power in eight hours. This sequence represents the SWS portions of the rebaselined Zion Review sequence no.4 and no.6. of Table A.4	1.5E-7	---	SE
-	Loss of main feedwater, AFW failure, failure of feed and bleed cooling	---	1.4E-7	TEFC
13	Same as sequence 12 above, only this is the CCW portion of the rebaselined Zion Review sequence no. 4 of Table A.4	1.0E-7	---	SEC
14	Interfacing systems LOCA	1.0E-7	1.1E-7	V
15	Failure of dc Bus 112, causing loss of one PORV and loss of ac Bus 148, failure of AFW	5.0E-8	---	TEFC
	Total	1.5E-4	3.2E-5	

*Not identified or provided as dominant sequences.

Table A.6 Comparison of Dominant Sequences for PWRS With Large-Volume Containments

Event Type	Core-Damage Frequency (per reactor year) NUREG-1150 Zion	IDCOR Zion	NUREG-1150 Surry	Millstone-3[21]	Oconee[22]
Loss of CCW or SW	5.5×10^{-6}*	--	--	7.1×10^{-7}	1.3×10^{-5}
Transients	--	--	--	9.3×10^{-6}	1.4×10^{-5}
LOCA**	2.7×10^{-5}	2.8×10^{-5}	7.0×10^{-6}	9.4×10^{-6}	1.6×10^{-5}
Loss of Bus	--	--	5.0×10^{-6}	1.0×10^{-5}	--
Station Blackout	3.0×10^{-6}	2.0×10^{-7}	9.5×10^{-6}	4.0×10^{-6}	--
Loss-of-Offsite Power	1.0×10^{-6}	2.2×10^{-6}	1.1×10^{-6}	3.2×10^{-6}	2.4×10^{-6}
Interfacing LOCA	1.1×10^{-7}	1.1×10^{-7}	9.0×10^{-7}	1.9×10^{-6}	2.8×10^{-6}
ATWS	$<1.0\times10^{-8}$	$<1.0\times10^{-8}$	1.6×10^{-6}	2.5×10^{-6}	6.0×10^{-6}
Internal Flooding	--	--	--	--	8.8×10^{-5}
Other	5.0×10^{-8}	1.6×10^{-6}	--	4.9×10^{-6}	--
Total Core-Damage Frequency	5.5×10^{-5}*	3.2×10^{-5}	2.6×10^{-5}	4.6×10^{-5}	1.4×10^{-4}

*Revised based on BNL SARRP results (NUREG/CR-4551, Vol. 5).
**Includes reactor-vessel rupture.
***Includes steam-generator tube rupture.

Table A.7 Review of Accident Analyses for Zion Nuclear Plant

Detailed Calculations			MARCH-CORRAL-CRAC Analysis				Baseline
IDCOR	SARP	SARRP BNL - Rebaseline[11]	ZPSS[5]	ZPSS Review[3,3] (SNL, BNL)	BNL[13]	NRC[12]	IDCOP[1]
Sequence Analyzed	Sequence Analyzed	Sequence Analyzed	Sequence Analyzed*	Sequence Analyzed	Sequence Analyzed	Sequence Analyzed	Sequence Analyzed
2-SE (TMLB'-β)[1,16] seal LOCA, containment bypass, or (S_2D β-seal LOCA)		11-SE (TMLB'γ,α)	1-SLFC 2-SE 3-ALFC 4-ALFC	1-SEFC 1-SEFC 3-SEFC 4-SEC	(TMLB')	(TMLB'α)	1-SEFC 2-SE 3-ALFC 4-ALFC
2-SE (TMLB'-δ)[1,14] seal LOCA or (S_2D-seal LOCA)	2-SE (S_2DC_r) seal LOCA[10] 2-SEγ ($S_2DC_{ir}F_{ir}$ 1) seal LOCA[10] 2-SE ($S_2DC_{ir}F_{ir}$2) seal LOCA[10]	1-SEFC (S_2D, seal LOCA)	5-ATWS, FW loss 6-ATWS, Turb loss 7-SLFC	5-SLF 6-SEFC 7-TEFC	(S_1D) (S_1B) (S_1B+one spray)	($S_2D\alpha$)	5-TEFC 6-TEC 7-SLFC
			8-SLC 9-AEFC	8-SE 9-ALF	(S_1B+flooded cavity) (S_1B+break size)		8-SLC 9-AEFC
14-TE (TMLB'-δ)[1,14] no LOCA	10-AEFC (S_2D-ε)[9] 14-TE (TMLB')-ε[9] 14-TE (TMLU)-γ[10]	4-SEC (TMLB'γ) TEFC (TMLB'δ)	10-AEFC 11-TEC [illegible]-TEC 13-TEC 14-TE (TMLB')	10-ALF 11-SE 12-AEFC 13-TEFC 14-TEFC	(TMLB') (TMLB'-flooded cavity) (TMLB'+basalt concrete) (TMLB'+sand)	(TMLB'ε) (TMLB'δ_2)	10-AEFC 11-TEC 12-TEC 13-TEC 14-TE (TMLB')
16-VL[1,16]		15-V	15-TEFC 16-VL	15-V	(TMLB'+spray) ([illegible]ML) (TMLQ)		15-TEFC 16-VL 17-AE 18-SLF
1-SLFC ($S_2H\delta$)[1,14]		5-SLF ($S_2H\delta$)			(S_1HF) (S:HFX fan failure)	(S_2HF)	
3-ALFC(AHδ)[1,14]		9-ALF (AH)					
4-ALFC (AHδ)[1]		12-AEFC (AD)					

Table A.8 Comparison of Conditional Early Containment Failure Probability Given a Core-Melt Event

	5th	95th	Median	Mean
Zion[11] (Revised 12/8/86)	1.0×10^{-3}	0.17	.01	.04
Surry*	1.5×10^{-2}	0.50	0.1	0.2
RSS*	-	-	-	0.2
IDCOR*	-	-	-	5×10^{-3}

*Data from the Surry Report.[20]

Table A.9 Results of the Integrated Analysis of Accident Process and Fission-Product Transport Zion

	Sequence*	Core Uncovery Time (hrs)	Start of Core Melt (hrs)	Time of RPV Failure (hrs)	Time of Containment Failure (hrs)	Fraction of Cladding Reacted	Maximal Containment Pressure (psi)	- (hrs)
1.	2-SE (TMLB')-β[1,14] (seal LOCA) - IDCOR	2.2	2.9	3.8	0	NR	NR	NR
2.	2-SE (S_2DC_r)[10] (seal LOCA) - SARP	1.08	1.57	2.22	24.0	0.47	149.0	24.0
	2-SE (TMLB')-δ[1,14] (seal LOCA) - IDCOR	2.2	3.0	3.8	32	0.153[a]	149.0	32.0
3.	2-SEγ ($S_2DC_{ir}F_{ir}$ 1)[10] (seal LOCA) - SARP	1.08	1.57	2.21	2.22	0.47	149.0	2.22
4.	2-SE ($S_2DC_{ir}F_{ir}$ 2)[10] (seal LOCA) - SARP	1.08	1.57	2.21	14.93	0.47	149.0	14.93
5.	14-TE (TMLB')-δ[1,14] (no seal LOCA) - IDCOR	2.2	3.1	4.0	32.0	0.143[a]	149.0	32.0
6.	14-TE (TMLB')-ε[9] - SARP	1.83	2.18	2.83	NCF(10h)	0.51	114.1	16.7
7.	14-TE (TMLU)-γ[10] - SARP	2.08	2.47	3.16	3.16	0.52	149.0	3.16
8.	16-VL[1,14] - IDCOR	~24	~24	26	0	NR	NR	NR
9.	1-SLFC (S_2H)-δ[1,14] - IDCOR	7.2	12	13.9	NCF	0.655[a]	34.0	13.9
10.	3-ALFC (AH)-δ[1,14] - IDCOR	0.8	1.7	2.3	NCF	0.486[a]	34.0	3.5
11.	10-AEFC (S_2D)-ε[9] - IDCOR	1.89	2.51	3.13	NCF(10h)	0.85	25.4	3.13

*Symbols are defined in Table A.14.
NCF = No containment failure.
NR = Not reported in Ref. 1.
[a]Calculated from amount of H_2 at RPV failure, given in Ref. 14.

Table A.10 SARP - IDCOR Calculated Source Terms

Reference	Sequencexxx	Puff*	Time of Release (hr)	Duration* of Release (hr)	Fractional Release								
					Xe	I	Cs	Te	Sr	Ru	La	Ce	Ba
SARP[10]	S_2DC_r (seal LOCA)	1	24	0.5	0.98	2.9E-6	1.2E-5	1.2E-3	6.2E-4	1.4E-8	3.4E-6	9E-6	5E-4
		2	25	3.0	0	0	4E-6	7.2E-4	2.8E-4	1.4E-8	1.5E-6	4E-6	2.3E-4
SARP[10]	$S_2DC_{F\vert r\vert r}^{1}$ (seal LOCA)	1	2.2	0.5	0.97	0.18	0.18	0.14	9.4E-5	1.1E-7	9.8E-9	0	2E-3
		2	3.5	5.5	0	0.04	0.04	0.18	3.7E-2	1.7E-4	1.8E-3	6.4E-4	2.6E-2
SARP[10]	$S_2DC_{F\vert r\vert r}^{2}$ (seal LOCA)	1	15	0.5	0.97	0.012	0.012	0.077	4.2E-3	2.8E-4	2E-4	7.2E-5	3.2E-3
		2	16	2.0	0	4E-3	4E-3	0.026	1.4E-3	9E-5	7E-5	2.6E-5	1E-3
SARP[10]	TMLU	1	3.2	0.5	0.99	4.9E-3	5.4E-3	0.025	7.6E-5	1.3E-7	9.9E-9	0	1.4E-3
		2	4.5	5.5	0	8E-4	1E-3	0.015	1.8E-5	3E-7	3.8E-7	1.9E-8	4E-4
SARP[9]	TMLB'-ε	-	3.2	-	0.98	2.1E-6	2.1E-6	8.4E-5	3.1E-5	5.9E-7	1.8E-6	1.6E-6	3.1E-5
SARP[9]	S_2D	-	NCF	-	0.98	2.5E-8	2.3E-8	3.6E-8	VS	VS	VS	VS	VS
IDCOR[14]	TMLB'-δ (seal LOCA)	-	32	-	0.98	1.7E-3	1.7E-3	2.0E-5	<1.0E-5	<1.0E-5	VS	VS	<1.0E-5
IDCOR[16]	TMLB'-δ (no LOCA)	-	32	-	0.98	1.7E-3	1.7E-3	2.0E-5	<1.0E-5	<1.0E-5	VS	VS	<1.0E-5
IDCOR[16]	:6-VL	-	0	-	0.98	8.0E-5	8.0E-5	8.0E-5	VS	VS	VS	VS	VS
IDCOR[14]	1-SLFC	-	NCF	-	VS	VS	VS	VS	VS	VS	VS	VS	VS
IDCOR[14]	3-ALFC	-	NCF	-	VS	VS	VS	VS	VS	VS	VS	VS	VS
IDCOR[16]	TMLB'-β (seal LOCA)	-	0	-	0.98	0.01	0.01	3.0E-4	VS	VS	VS	VS	VS

*The distribution of fractional releases between two puffs and duration of release are reprinted from Ref. 11.
NCF = no containment failure.
VS = very small (negligible)

Table A.11 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 Ex-Vessel																			
Yes		•		•				•											
No	✓		✓		✓	✓	✓		•							✓	✓	✓	✓
Direct Heating Effect																			
None	✓	✓	✓	✓															
Significant																✓	✓	✓	✓

✓ - The characteristic is required for the bin. Characteristics not marked are not determinant of the bin and any combination may apply.

• - Either of the characteristics noted by this symbol apply to the bin in combination with the others marked ✓.

Note 1: The spray question is also dependent on timing. Critical timeframes 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.

Table A.12 Results of Source Term Calculations Release Fraction By Group

Bin	Release Time (hr)	Dur. (hr)	Warning Time (hr)	Elevation (m)	Energy BTU/hr	1 KR-XE	2[a] I	3[b] CS	4 TE	5 SR	6 RU	7 LA	8 CE	9 BA
1	2.5	5.	0.5	10	2.+7	.88 ±.21	.59 ±.19	.43 ±.22	.40 ±.21	5.1-2 ±4.9-2	4.5-3 ±8.5-3	1.4-2 ±2.4-2	3.3-3 ±5.5-3	5.9-2 ±5.3-2
2	2.5	5.	0.5	10	8.+6	.88 ±.21	.23 ±.051	.14 ±.07	8.2-2 ±6.4-2	1.1-3 ±1.2-3	1.1-6 ±1.9-6	1.4-4 ±3.5-4	6.1-5 ±1.4-4	5.0-3 ±5.7-3
3	2.5	5.	0.5	10	2.+7	.88 ±.21	9.1-2 ±6.1-2	3.6-2 ±4.2-2	.18 ±.11	5.0-2 ±4.8-2	4.5-3 ±8.5-3	1.4-2 ±2.4-2	3.3-3 ±5.5-3	4.6-2 ±5.0-2
4	2.5	5.	0.5	10	8.+6	.88 ±.21	8.9-2 ±6.1-2	3.4-2 ±4.2-2	1.6-2 ±2.6-2	8.2-4 ±1.1-3	7.0-7 ±1.8-6	1.4-4 ±2.4-4	6.1-5 ±1.5-4	4.9-4 ±5.3-3
5	1.5	5.	1.0	10	1.+7	.88 ±.21	.43 ±.18	.37 ±.18	.36 ±.19	5.1-2 ±4.8-2	4.5-3 ±8.5-3	1.4-2 ±2.4-2	3.3-3 ±5.5-3	6.0-2 ±5.2-2
6	2.5	5.	0.0	10	1.+7	.88 ±.21	.12 ±.08	8.8-2 ±.10	.17 ±.11	5.0-2 ±4.8-2	4.5-3 ±8.5-3	1.4-2 ±2.4-2	3.3-3 ±5.5-3	4.6-2 ±5.0-2
7	1.5	5.	0.0	10	1.+5	.88 ±.21	8.9-2 ±6.1-2	3.4-2 ±4.2-2	1.6-2 ±2.6-2	4.1-4 ±8.5-4	8.1-7 ±3.1-6	8.3-5 ±3.0-4	3.0-5 ±1.0-4	6.9-4 ±1.1-3
8	8.0	1.	6.0	10	1.+6	.88 ±.21	5.3-2 ±3.8-2	9.6-3 ±1.7-2	1.3-3 ±1.4-3	3.9-4 ±8.5-4	7.9-7 ±3.1-6	8.3-5 ±3.0-4	3.0-5 ±1.0-4	2.2-4 ±1.1-3
9	15.0	1.	12.0	10	1.+6	.88 ±.21	.18 ±.12	.13 ±.13	.14 ±.13	1.1-2 ±1.5-2	1.8-3 ±4.8-3	2.9-3 ±6.8-3	6.1-4 ±1.4-3	1.2-2 ±1.5-2
10	2.5	5.	0.5	10	2.+5	.88 ±.21	.18 ±.12	.13 ±.13	5.7-2 ±5.7-2	2.3-3 ±2.5-3	3.9-4 ±9.7-4	6.0-4 ±1.3-3	1.3-4 ±2.6-4	4.4-3 ±4.8-3

a. Late iodine and revolatilization releases are included.

b. Revolatilized Cs release is included.

c. If the lower error limit reaches a negative value for release fraction, then lower limit is zero.

Table A.12 (Continued)

Bin	Release Time (hr)	Dur. (hr)	Warning Time (hr)	Elevation (m)	Energy BTU/hr	1 KR-XE	2[a] I	3[b] CS	4 TE	5 SR	6 RU	7 LA	8 CE	9 BA
11	2.0	5.	1.0	0	3.+6	.88 ±.21	.13 ±.17	6.2-2 ±.15	6.0-2 ±7.9-2	1.1-2 ±1.9-2	6.9-4 ±2.7-3	2.2-3 ±7.7-3	5.6-4 ±1.8-3	1.1-2 ±1.9-2
12	2.0	5.	1.0	0	3.+6	.88 ±.21	.42 ±.19	.27 ±.18	.18 ±.132	2.5-2 ±2.9-2	4.9-3 ±4.6-3	6.0-3 ±1.3-2	1.5-3 ±3.0-3	2.9-2 ±3.1-2
13	24.0	1.	22.0	0	0	.88 ±.21	2.1-5 ±2.0-5	2.1-5 ±2.0-5	3.9-5 ±3.3-5	4.1-6 ±4.8-6	7.5-7 ±1.8-6	1.2-6 ±2.5-6	2.4-7 ±4.9-7	4.6-6 ±5.2-6
14	24.0	1.	22.0	0	0	.88 ±.21	2.8-6 ±4.1-6	2.9-6 ±4.2-6	2.6-6 ±3.9-6	1.4-7 ±1.9-7	2.7-10 ±7.0-10	2.8-8 ±6.8-8	1.0-8 ±2.2-8	2.0-7 ±3.3-7
15	24.0	1.	22.0	10	0	.88 ±.21	2.8-6 ±4.1-6	2.9-6 ±4.1-6	2.6-6 ±3.3-6	1.4-7 ±1.9-7	2.7-10 ±7.0-10	2.8-8 ±6.8-8	1.0-8 ±2.2-8	2.0-7 ±3.3-7
16	1.5	5.	0.5	10	2.+7	.95 ±.09	.65 ±.14	.49 ±.19	.51 ±.13	8.2-2 ±5.0-2	.10 ± .12	2.8-2 ±2.[illegible]-2	3.7-3 ±3.1-3	9.1-2 ±5.2-2
17	1.5	5.	0.5	10	8.+6	.93 ±.22	.28 ±.07	.19 ±.08	.23 ±.07	4.4-2 ±3.1-2	7.3-2 ±8.5-2	1.6-2 ±1.2-2	1.7-3 ±1.2-3	4.7-2 ±3.1-2
18	2.5	5.	0.5	10	2.+7	.92 ±.15	.12 ±.08	7.0-2 ±6.9-2	.23 ±.11	6.6-2 ±4.3-2	5.5-2 ±6.0-2	2.0-2 ±2.0-2	3.5-3 ±4.1-3	6.3-2 0E+00
19	2.5	5.	0.5	10	8.+6	.92 ±.15	.12 ±.08	6.6-2 ±5.6-2	6.0-2 ±3.4-2	1.3-2 ±8.5-3	1.9-2 ±2.0-2	5.6-3 ±5.5-3	6.2-4 ±5.7-4	1.3-2 ±8.2-3

a. Late iodine and revolatilization releases are included.

b. Revolatilized Cs release is included.

c. If the lower error limit reaches a negative value for release fraction, then lower limit is zero.

Table A.13 NRC/IDCOR Issues

Issue	Subject
1	Fission-Product Release Prior to Vessel Falure
2	Recirculation of Coolant in Reactor Vessel
3	Release Model of Control Rod Materials
4	Fission Product and Aerosol Retention in the Primary System
5	In-Vessel H_2 Generation
6	Core Slump, Core Collapse, and Reactor Vessel Failure
7	Containment Failure due to In-vessel Steam Explosions
8	Direct Heating of Containment
9	Ex-Vessel Fission-Product Release
10	Ex-Vessel Heat Transfer Model from Molten Core to Concrete
11	Revaporization of Fission Products from the Primary System
12	Fission Product Deposition Model in Containment
13A	Suppression Pool Bypass (Pool Scrubbing)
13B	Retention of Fission Products in Ice Beds
14	Modeling of Emergency Response
15	Containment Performance
16	Secondary Containment Performance
17	Hydrogen Ignition and Burning
18	Essential Equipment Performance

Table A.14 List of Symbols for Reactor Accidents

A	Intermediate to large LOCA.
B	Failure of electric power to ESF's.
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, T_1.
D	Failure of the emergency core cooling injection system.
F	Failure of the containment spray recirculation system.
F_1, F_2, F_3	Different variants of F.
H	Failure of the emergency core cooling recirculation 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.
S_1	A small LOCA with an equivalent diameter of about 2-6 inches.
S_2	A small LOCA with an equivalent diameter of 1/2-2 inches.
S_3	Small accident with an equivalent diameter 0.75 inch = pump seal LOCA.
SGTR	Steam generator tube rupture.
T	Transient event.
T_{23}	Transient event other than T_1 = transient from loss of offsite power.
U	Chemical and volume control system.
V	LPIS check valve failure.
W	Failure to remove residual core heat.
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.
θ	Water present in cavity at RPV failure time.

Table A.15 Risk Results from the SARRP Rebaselining Report Compared to Previous Studies

Index (Per Year)	Mean	Median	5%	95%	RSS (Surry)	ZPSS[5]	BNL[7] Review	IDCOR[1]
1. Early Deaths	2.3-4*	6.5-5	3.5-6	8.3-4	4.-5	1.1-5	6.5-5	0.0
2. Early Illness	5.4-4	1.6-4	1.605	1.9-3	--	2.8-4	3.5-3	--
3. Cancer Deaths	3.9-2	1.7-2	3.0-3	1.2-1	3.-2	3.6-3	1.6-2	1.6-5
4. Off-site Costs	1.5+4	6.5+3	8.0+2	4.3+4	--	--†	--	--
5. Population Dose	8.4+1	4.5+1	7.3+0	2.3+2	--	1.5+1	2.2+2	.31

*2.3-4 = 2.3×10^{-4}.

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PWR, ICE-CONDENSER CONTAINMENT DESIGN

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ABSTRACT

Guidelines and criteria have been developed for preventing and mitigating severe accidents in PWRs that have ice-condenser containments. The guidelines were developed from insights derived from reviews of risk assessments performed specifically for the Sequoyah plant and from assessments of other relevant studies. Accident sequences that dominate the core-damage frequency and those accident sequences that are of potentially high consequence were identified. Vulnerabilities of the ice-condenser containment to severe accident containment loads were also identified. In addition, those features of a PWR with an ice-condenser containment, which are important for preventing core damage and are available for mitigating fission-product release to the environment were also identified. The guidelines and criteria are issued to provide direction to an analyst examining an individual plant. This direction calls attention to plant features and operator actions and provides the standards for assessing those features and actions found to be helpful in reducing the overall risk for Sequoyah and other PWRs with ice-condenser containments. Thus, the guidance is offered as a resource in examining the subject plant to determine if the same, or similar, guidelines will be of value in reducing overall plant risk. These guidelines and criteria are intended to serve solely as guidance.

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ACKNOWLEDGMENTS

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NOMENCLATURE

ac	alternating current
AFW	auxiliary feedwater
AFWS	auxiliary feedwater system
ASEP	Accident Sequence Evaluation Program
ATWS	anticipated transients without scram
BCL	Battelle Columbus Laboratories
BNL	Brookhaven National Laboratory
BWR	boiling water reactor
CCW	component cooling water
CCWS	component cooling water system
CDF	core-damage frequency
CHR	containment heat removal
CLAS	cold leg accumulator system
CSRS	containment spray recirculation system
CSS	containment spray system
dc	direct current
ECC	emergency core cooling
ECCS	emergency core cooling system
ERG	Emergency Response Guidelines
ESS	engineering safety systems
ESW	emergency service water
FSAR	Final Safety Analysis Report
GI	generic issue
HPI	high-pressure injection
HPIS	high-pressure injection systems
HPRS	high-pressure recirculation system
ICS	ice-condenser system
IDCOR	Industry Degraded Core Rulemaking Program
IPE	individual plant examination
ISL	interfacing system LOCA
LLRT	local leak rate testing
LOCA	loss-of-coolant accident
LIP	low-pressure injection
LPIS	low-pressure injection system
LPRS	low-pressure recirculation system
LWR	light-water-reactor
MFW	main feedwater
MSIV	main steam isolation valve
MW	megawatt
NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
NRC/RES	U.S. Nuclear Regulatory Commmission, Office of Nuclear Regulatory Research
NRR	Office of Nuclear Reactor Regulation
PORV	power operated relief valve
PRA	probabilistic risk assessment
PWR	pressurized water reactor
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RPS	reactor protection system
RPV	reactor pressure vessel

NOMENCLATURE (Cont'd)

RSS	Reactor Safety Study
RSSMAP	Reactor Safety Study Methodology Applications Program
RWST	refueling water storage tank
SARP	Severe Accident Research Program
SARRP	Severe Accident Risk Reduction Program
SBO	station blackout
SG	steam generator
SGTR	steam generator tube rupture
SI	safety injection
SNL	Sandia National Laboratories
SW	service water
SWS	service water system
UHIS	upper head injection system
USI	unresolved safety issue

1. EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) has formulated an approach for a systematic safety examination of existing plants to determine whether particular severe accident vulnerabilities are present and what changes are desirable to ensure that there is no undue risk to public health and safety.

The Industry Degraded Core Rulemaking Program (IDCOR) selected four reference plants for detailed analysis: Peach Bottom, Grand Gulf, Sequoyah, and Zion. The IDCOR analyses performed for the reference plants have been documented together with the methodology used for the analyses and the technical basis supporting the methodology.

Parallel with the IDCOR work, the NRC under the Severe Accident Research Program (SARP), performed risk assessments, audit calculations, sensitivity studies, and uncertainty analyses for five plants. The five plants considered by SARP were Peach Bottom, Grand Gulf, Sequoyah, Zion and Surry.

The purpose of this effort is to review all of the IDCOR and SARP analyses performed for the reference plants, understand the reasons for the differences, and then use the experience gained from these reviews for developing guidelines and criteria that identify plant features and operator actions that were found to be important for either preventing or mitigating severe accidents in each plant type. In turn, these guidelines should prove helpful in the systematic safety examination of individual plants.

Three basic objectives or goals for this severe accident program apply equally to all plant types:

- Goal 1: Mitigate fission-product releases.
- Goal 2: Control the frequency of high-consequence sequences.
- Goal 3: Reduce high core-damage frequency.

The aim was, therefore, to develop detailed guidelines and criteria that could be used to achieve these goals during the examination of individual plants.

"Guidelines," as used in this report, identify those plant features and operator actions that were found to be important to either preventing or mitigating severe accidents in the reference plant studies. These guidelines provide a list of plant features and operator actions that the utilities can use as part of each individual plant examination (IPE). It is not the intent of this report to specify a set of improvements for either the reference plant or for any other plant which would be sufficient to achieve a certain level of safety. Instead, the guidelines indicate potential improvements in various areas of plant design and operation of which each utility should be aware when conducting its IPE and making decisions on plant improvements. The intent is to provide guidance to the analyst performing an IPE, as to the plant features and operator actions which were found to reduce overall risk. It is prudent to check whether potential improvements identified in studies of other similar plants can be of help in improving overall plant performance. The guidelines contained in this report, therefore, complement the IPEs.

"Criteria," as used in this report, are the attributes which have been identified as important to assess the performance of plant features and operator actions identified in the guidelines. The criteria provide deterministic (as opposed to probabilistic) performance measures which are judged to be helpful to implement the concepts contained in the guidelines. When a decision is made to provide an item or an action specified in a guideline, the utility should address a set of estions relating to the design, operation and availability of the needed equipment and the training of operators. For the example of containment spray guidance, the capacity of the spray system, the selection of setpoints to initiate sprays, the volume of the water source, the availability of applicable procedures and the accessibility of certain valves by operators should be addressed. The criteria on containment sprays provide helpful information in assessing the mitigative capability of the containment spray system in each individual plant.

Based on an extensive review of prior severe accident investigations, the authors have provided a set of guidelines and associated criteria which can be used to assess the capability of individual pressurized water reactor (PWR) plants with ice-condenser containments, to cope with severe accidents. Although much of the work is based on probabilistic risk assessments (PRAs), the guidelines and criteria are deterministic in nature. That is the criteria describe specific features of key systems and operational procedures which have been found helpful in reducing the likelihood of severe accidents. The guidelines and criteria take into account detailed severe accident experiments and analyses performed by the NRC/RES, the nuclear power industry and foreign governments.

The following sections present the insights gained from reviewing the PRAs. Specifically, the IDCOR Sequoyah Integrated Containment Analyses[1] and the SARP Sequoyah reports[2-4] were reviewed in detail. These studies were compared with the original Sequoyah risk assessment in the Reactor Safety Study Methodology Applications Program (RSSMAP)[5] and relevant PWR PRAs for other plants, including Oconee,[6] Zion,[7] Surry[8] and Millstone-3.[9]

1.1 Core-Damage Profile

PRAs for PWRs with ice-condenser containments have indicated that transients (including loss-of-component cooling water (CCW) and station blackout (SBO) and small loss-of-coolant accidents (LOCAs) (including pump seal LOCA and interfacing systems LOCA) tend to dominate the risk profile. There was no consistent pattern of relative ranking of transient sequences across all of the studies. However, in the IDCOR and Accident Sequence Evaluation Program (ASEP) studies the same few functional accident sequences figured prominently in the core-damage frequency (CDF) profiles.

For the RSSMAP study of Sequoyah, accidents involving loss of primary system coolant makeup following LOCA appeared as dominant contributors (about 85%) to the CDF. Although the IDCOR and ASEP studies have indicated reduced frequencies for these sequences, criteria have been developed to ensure that the frequency of these sequences is controlled for other PWR ice-condenser plants.

In addition to the LOCA sequences, ASEP indicated that sequences involving SBO and sequences involving CCW failures are the dominant core-damage

sequences for Sequoyah. The sequences initiated by loss of CCW are dominated by common cause events and were not identified by IDCOR.

1.2 Consequence Analysis

The assessment of core-meltdown phenomena and containment response in the available PRAs indicated that the ice-condenser containment is vulnerable to overpressurization because of the buildup of noncondensable gases because of its relatively small volume and low design pressure. Under some conditions the NRC Severe Accident Risk Reduction Program (SARRP) predicts that the containment has the potential to fail a short time (a few hours or less) after the reactor vessel fails. ASEP/SARRP predicts that the fission-products released into the containment will nevertheless be substantially reduced by deposition in the ice. Thus, even with a containment failure, some containment function (reducing the source term) is preserved for almost all cases. Only direct bypass sequences (interfacing systems LOCA (ISL)) result in severe releases of fission products. However, SARRP predicts eventual containment failure for most sequences and the overall impact appears to be a substantially higher fission-product releases than that calculated by IDCOR.

1.3 Guidelines and Criteria

The guidelines have been developed to translate the three goals of the severe accident program into deterministic criteria for assessing each plant's response for the dominant types of core-damage sequences.

Each guideline is provided with a detailed list of criteria which provide helpful information to assess the performance of plant features and operator actions identified in the guidelines.

1.3.1 Mitigate Fission-Product Releases

For a PWR with an ice-condenser containment, the dominant core-damage sequences were found to be small breaks (including ISL and pump seal LOCA) and transients (including loss of CCW and SBO). In order to minimize off-site consequences, the containment systems must be able to fulfill their role of restricting fission-product releases even under severe accident conditions. Three guidelines and associated criteria have been developed which emphasize the importance of maintaining containment function for the threats to containment posed by these dominant sequences.

Guideline 1 - Maintain Containment Integrity

Pressure buildup because of noncondensable gas generation and hydrogen combustion during a core-meltdown accident can threaten the ice-condenser containment. Both IDCOR and ASEP/SARRP analyses concluded that for some cases containment failure will eventually occur unless mitigative actions are taken. If containment failure occurs, significant fission-product release from the containment atmosphere to the environment could occur. This guideline and associated criteria identify those systems which will delay or prevent overpressure failure.

Guideline 2 - Ice-Condenser System (ICS) and Air Return Fan System (ARFS)

In the Sequoyah plant, the ICS plays an essential role in condensing steam and scrubbing fission products for a broad range of severe accidents. In order to preserve these functions, it is essential that ice-condenser bypass conditions be avoided. The ARFS helps the ICS and the containment spray system by circulating air between the lower and upper containment compartments through the ice condenser. The ARFS also promotes containment mixing and a more uniform concentration of hydrogen in the containment thus preventing high concentration regions from occurring and allowing hydrogen combustion to occur at low concentrations which do not threaten the containment.

Guideline 3 - Hydrogen Control

The most severe consequences of core-melt accidents in Sequoyah are from accidents which result in early containment failure. The SARRP analyses indicates that hydrogen ignition at high concentrations may cause such early failure. In order to prevent such uncontrolled ignition, both the ARFS and the hydrogen igniters must be operable. The igniters continuously burn the hydrogen at low concentrations in the containment air while the fans promote uniform mixtures. Such controlled hydrogen burns will avoid detonation or deflagration and containment integrity will be preserved during hydrogen burning.

1.3.2 Control the Frequency of High-Consequence Sequences

Guideline 4 - Interfacing Systems LOCA (Including Steam Generator Tube Rup-Rupture)

Although the interfacing systems LOCA (ISL) is usually a highly unlikely event, it could be a significant risk contributor because of the potentially high release. Although steam generator tube rupture (SGTR) with one or more tubes ruptured bears the characteristic of a small LOCA, it is unique in the sense that it is also a potential containment bypass LOCA, releasing reactor coolant into the secondary side of the steam generators (SG). Thus the SGTR provides several potential paths for release of fission product to the environment outside the containment via the main steamline, turbine, turbine bypass, condenser, condenser exhaust, SG atmospheric steam dump valves or safety valves and the SG blowdown line. Thus, despite its relatively small CDF (usually a few percent of the overall CDF from internal events), Guideline 4 has been developed to prevent the occurrence of SGTR and ISL events or to mitigate the potentially high fission-product releases if one occurs.

1.3.3 Reduce High Core-Damage Frequency Sequences

The major contributors to the core damage for Sequoyah have been identified as small LOCA (including ISL and pump seal LOCA), SBO and loss of CCW transients. Thus, if the frequencies of these accident sequences (or subset of them) can be reduced, then the overall CDF could be substantially reduced.

Guideline 5 - Operator Response for Recirculation

The small LOCA initiator (S_2) has been identified as the largest contributor to CDF at Sequoyah. Major contributors to the S_2 sequences are operator failures in the recirculation switchover operation and common cause failures

of the containment sump suction valves and the high-pressure pump suction valves. Although the sequences involving loss of containment heat removal (CHR) were found only marginally important for Sequoyah, the main contributor to these sequences is also the failure in the recirculation switchover. Thus, Guideline 5 and the associated criteria have been developed to ensure a low frequency of recirculation failure.

Guideline 6 - Station Blackout

In Sequoyah, station blackout (SBO) sequences are important contributors to CDF. The accident sequences in SBO are characterized by two categories of events: First, ac power is not recovered before battery depletion which, in turn, defeats the turbine-driven auxiliary feedwater system (AFWS) pump, and second, SBO causes a reactor coolant pump (RCP) seal LOCA because of failures of seal injection flow and RCP thermal barriers resulting from loss of the CCW and service water systems.

The AFWS is the normal means of decay heat removal in small LOCA and transient events, including a normal plant shutdown. The previous studies have indicated that the core-damage sequences at Sequoyah involving failure of the AFWS are related to (1) the SBO controlled by failure of the AFWS turbine-driven pump because of dc bus failure or battery depletion, and (2) the loss of a dc bus initiator followed and controlled by failures of the motor-driven pumps and by failure of the reactor coolant system (RCS) feed & bleed cooling.

For accidents involving the loss-of-offsite power and onsite emergency power, the NRC recommends examining the proposed SBO rule for applicability. The criteria associated with Guideline 6 are intended to emphasize the need to search for plant specific features and potential common cause failures which could disable systems required to work during an SBO. For individual plants which are found to have a vulnerability to SBO, the criteria highlight the importance of proper emergency procedures and operator training in recovering from an SBO event.

Guideline 7 - Component Cooling Water

An event initiated by loss of component cooling water (CCW) may lead to RCP seal LOCA sequences because of loss of RCP seal cooling, under the condition that high-pressure injection (HPI) fails because pumps in the high-pressure injection systems (HPIS) are also cooled by CCW. The contribution of this initiator to CDF can be reduced by reducing the frequency of loss of the CCW system or by decreasing the dependence of RCP seals on the CCW.

Guideline 8 - Reactor Coolant System (RCS) Depressurization by Secondary Blowdown

One potentially effective means of making up core inventory in case the high-pressure systems are unavailable, either in injection or in recirculation phase, is via secondary blowdown, that is to depressurize the reactor coolant system (RCS) by heat removal through the steam generators (SG). After sufficient heat removal, the low-pressure injection system (LPIS) can be actuated for the RCS makeup operation. To attain success, the operator must open the SG atmospheric steam dump valves, maintain AFW or main feedwater to the SG and have one of the LPIS trains available for RCS makeup.

Guideline 9 - Reactor Coolant System (RCS) Feed & Bleed Cooling

A second potentially effective means of making up core inventory in case the HPIS are unavailable, either in the injection or in the recirculation phase, is to depressurize the RCS by the power operated relief valves (PORVs). After depressurization, the LPIS can be actuated for the makeup operation. This emergency procedure appears to have a significant impact on reducing the CDF. To attain success in this procedure, the operator must open the PORVs and have one of the LPIS trains available for RCS makeup.

Guideline 10 - Anticipated Transients Without Scram (ATWS)

ATWS has not been found to be a significant contributor to core damage at Sequoyah and this is also generally supported by PRAs for other Westinghouse PWRs. One factor for these results is the credit given to the manual scram and the emergency boration using the charging system to deliver borated water to the reactor vessel. Failures of the manual scram and the emergency boration are dominated by failures of operator actions.

The NRC promulgated the ATWS rule to reduce the frequency of ATWS events. The criteria developed for this guideline emphasize human reliability insights as to the importance of correct emergency procedures and operator training in recovering from an ATWS event.

Guideline 11 - Support System Interdependencies

Although the importance is difficult to quantify, one of the insights of most risk assessment studies is the importance of support system interdependencies. For example, a draft of the SARRP Peach Bottom study indicated that loss of all service water was a dominant contributor to core melt. The final version of the sequence studies have reduced it to one percent of the overall core melt. In order to ensure that support system vulnerabilities do not cause unacceptably high core-melt frequencies for other PWR, ice-condenser containment plants, Guideline 11 has been developed to help assess any weaknesses of the support systems.

Guideline 12 - Flooding of Emergency Equipment

Internal flooding has been found to be a significant contributor to core damage in only on PWR plant. However, the concerns appear to be of general applicability to other designs. Thus, this guideline was developed for PWR ice-condenser plants to assess the potential for flooding of safety-related equipment.

1.4 Using the Guidelines and Criteria

Numerous investigations, including PRAs, have been performed for the reference plants and similar plants by both the NRC and the nuclear power industry. The insights gained from many of the studies have been used in developing the guidelines and criteria contained in this report (including the other volumes of this report). These guidelines and criteria are issued to provide guidance to the analyst performing an IPE. This guidance is in the form of plant features, operator actions and the criteria for assessing those features and actions found to be helpful in reducing the overall risk for Sequoyah and

other PWR plants. Thus, the guidance is given to provide a resource in examining the subject plant to determine if the same, or similar, guidelines will be of value in reducing overall plant risk. These guidelines and criteria are intended to be used solely as guidance.

1.5 References for Section 1

1. "Sequoyah Nuclear Generating Station and Integrated Containment Analysis," IDCOR Technical Report T.23.1S, March 1985.

2. T. Wheeler, "Analysis of Core Damage Frequency from Internal Events: Sequoyah Power Sation, Unit 1," Sandia National Laboratories, NUREG/CR-4550, Volume 2, October 1986.

3. V. L. Behr et al., "Containment Event Analysis for Postulated Severe Accidents: Sequoyah Power Station, Unit 1," NUREG/CR-4700, Volume 2, Draft Report for Comment, February 1987.

4. A. S. Benjamin et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Sequoyah Power Station, Unit 1," NUREG/CR-4551, Draft Report for Comment, Volume 2, February 1987.

5. D. D. Carlson et al., "Reactor Safety Study Methodology Applications Program: Sequoyah #1 PWR Power Plant," Sandia National Laboratories, NUREG/CR-1659, Volume 1, April 1981.

6. W. R. Sugnet et al., "Oconee PRA - A Probabilistic Risk Assessment of Oconee Unit 3," Duke Power Co., NSAC-60, June 1984.

7. T. A. Wheeler, "Analysis of Core Damage Frequency from Internal Events: Zion Unit 1," Sandia National Laboratories, NUREG/CR-4550, Volume 7, October 1986.

8. R. C. Berturio et al., "Analysis of Core Damage Frequency from Internal Events: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4550, Volume 3, November 1986.

9. Northeast Utilities, "Millstone Unit 3 Probabilistic Safety Study," August 1983.

Table 1.1 Guidelines for Preventing and Mitigating Severe Accidents in a PWR with an Ice-Condenser Containment

Guideline		Description
		<u>Mitigate Fission Product Releases:</u>
1		Containment Integrity
	1.A.	Provide Containment Spray
	1.B.	Provide Filtered Venting
2		Ice Condenser System (ICS) and Air Return Fan System (ARFS)
	2.A.	Provide Reliable Operation of ICS and ARFS
3		Hydrogen Control
	3.A.	Provide Reliable Hydrogen Igniters
		<u>Control the Frequency of High-Consequence Sequences:</u>
4		Interfacing Systems LOCA (Including Steam Generator Tube Rupture)
	4.A.	Prevent Overpressurization of Low Pressure Systems
	4.B.	Prevent Steam Generator Tube Rupture and Minimize Its Consequences
		<u>Reduce High Core-Damage Frequency Sequences:</u>
5		Operator Response for Recirculation
	5.A.	Provide Adequate Recirculation Cooling
	5.B.	Provide Adequate Containment Heat Removal
6		Station Blackout
	6.A.	Provide Operator Response During Station Blackout
7		Component Cooling Water (CCW)
	7.A.	Provide Adequate Cooling to Engineered Safety Features Systems
8		Reactor Coolant System (RCS) Depressurization by Secondary Blowdown
	8.A.	Provide RCS Depressurization Capability
9		Reactor Coolant System (RCS) Feed & Bleed Cooling
	9.A.	Provide Operator Response for RCS Feed & Bleed Cooling
10		Anticipated Transients Without Scram (ATWS)
	10.A.	Provide Operator Response During ATWS
11		Support System Interdependencies
	11.A.	Examine Support System Interdependencies
12		Flooding of Emergency Equipment
	12.A.	Prevent or Mitigate Internal Flooding

2. INTRODUCTION

2.1 Background

The U.S. Nuclear Regulatory Commission (NRC) has formulated an approach for a systematic safety examination of existing plants to determine whether particular severe accident vulnerabilities are present and what changes are desirable to ensure that there is no undue risk to public health and safety.

The Industry Degraded Core Rulemaking Program (IDCOR) selected four reference plants for detailed analysis, namely:

- Peach Bottom (a BWR with a Mark I containment)
- Grand Gulf (a BWR with a Mark III containment)
- Zion (a PWR with a large dry containment)
- Sequoyah (a PWR with an ice-condenser containment)

The IDCOR analyses performed for the above reference plants have been documented together with the methodology used for the analyses and the technical basis supporting the methodology.

Parallel with the IDCOR work, the NRC under the Severe Accident Research Program (SARP), performed risk assessments, audit calculations, sensitivity studies, and uncertainty analyses for five plants. The five plants considered include the above four IDCOR reference plants, and, in addition

- Surry (a PWR with a subatmospheric containment)

The purpose of this effort is to review all of the IDCOR and SARP analyses performed for the reference plants, understand the reasons for the differences, and then use the experience gained from these reviews for developing guidelines and criteria that identify plant features and operator actions found to be important for either preventing or mitigating severe accidents in each plant type. In turn, these guidelines should be helpful in the systematic safety examination of individual plants.

The IDCOR Sequoyah analysis[1] was documented in March 1985 and was supplemented by additional sensitivity studies in July 1985. The SARP Sequoyah reports[2-4] were reviewed in draft form during 1986. These reports were published early in 1987 and were summarized in the "Reactor Risk Reference Document" (NUREG-1150),[5] which was published for comment in February 1987. The experience gained from the review of these Sequoyah studies along with other PWR PRA studies (namely, Zion, Surry, Oconee and Millstone-3) was used to generate the guidelines and criteria which are the subject of this report.

2.2 Objectives

Three basic objectives or goals for this severe accident program apply equally to all plant types:

- Goal 1: Mitigate fission-product releases.
- Goal 2: Control the frequency of high-consequence sequences.
- Goal 3: Reduce high core-damage frequency.

The aim was, therefore, to develop detailed guidelines and criteria that could be used to achieve these goals during the examination of individual plants. The guidelines and criteria are defined in the sections that follow.

2.2.1 Guidelines

"Guidelines," as used in this report, identify those plant features and operator actions that were found to be important to either preventing or mitigating severe accidents in the reference plant studies. These guidelines provide a list of plant features and operator actions that the utilities can use as part of each individual plant examination (IPE). It is not the intent of this report to specify a set of improvements for either the reference plant or for any other plant which would be sufficient to achieve a certain level of safety. Instead, the guidelines indicate potential improvements in various areas of plant design and operation of which each utility should be aware when conducting its IPE and making decisions on plant improvements. The intent is to provide guidance to the analyst performing an IPE, as to the plant features and operator actions which were found to reduce overall risk. It is prudent to check whether potential improvements identified in studies of other similar plants can be of help in improving overall plant performance. The guidelines contained in this report, therefore, complement the IPEs.

The three objectives or goals were noted as applying equally to all plant types. Although the goals are independent of plant type, the guidelines that are needed to achieve the goals are plant dependent. In general terms, Goal 1 implies that there should be effective means of mitigating the fission-product releases for the broad classes of accident sequences which dominate the core-damage frequency. Therefore, these dominant accident sequences have to be determined and those plant features and operator actions that are available to mitigate the release of fission products have to be identified. Only then can detailed guidelines be developed to ensure that these dominant accident sequences can be mitigated.

There may be accident sequences for which a specific plant will have substantial fission-product releases (e.g., containment bypass sequences). Thus, for such sequences Goal 1 may be difficult to achieve. Therefore, all reasonable steps are identified which could reduce the frequency of these potentially high-consequence sequences (namely Goal 2). Again, the accident sequences have to be identified and plant vulnerabilities and/or operator actions that lead to core damage for these sequences also have to be identified. Detailed guidelines can then be developed which will aid in assessing an individual plant's capability to prevent these sequences from occurring.

It is also desirable to ensure that the overall core-damage frequency is low (namely Goal 3). Again, the dominant accident sequences have to be found so that detailed guidelines can be developed to reduce the frequency of these sequences, if necessary.

In general, the following screening process was used to determine whether or not to develop a particular guideline:

- any accident sequence with a core-damage frequency greater than 10^{-6} per reactor year

- any sequence that contributed to more than 5% of the total core-damage frequency

- any event that caused a conditional probability of early containment failure greater than 0.1

- any sequence that resulted in containment bypass with a frequency greater than 10^{-7} per reactor year

- any sequence that was judged to be uniquely important (example, very severe consequences)

This screening process led to the development of guidelines that can be used in the systematic safety examination of other PWRs with ice-condenser containments. For example, the guideline for containment spray and long-term heat removal was identified as an item that would help to achieve Goal 1 (namely, to mitigate fission-product releases) for the PWR ice-condenser reference plant. Therefore, in the safety examination of other PWRs with ice-condenser containments, the capability for containment spray and long-term heat removal may need to be carefully assessed.

The development of a particular guideline for the PWR ice-condenser reference plant does not imply that this plant or any of the other plants in this category need to conform to this guideline. It simply means that analyses have indicated that this particular guideline has the potential to significantly reduce risk. Thus, the guidance is given to provide a resource in examining the subject plant to determine whether the same or similar guidelines will be of value in reducing overall plant risk. Whether or not the guideline is useful or needed in a particular PWR with an ice-condenser containment depends on plant-specific details and is beyond the scope of this report and is therefore not addressed here.

2.2.2 Criteria

"Criteria," as used in this report, are the attributes which have been identified as important to assess the performance of plant features and operator actions identified in the guidelines. The criteria provide deterministic (as opposed to probabilistic) performance measures which are judged to be helpful to implement the concepts contained in the guidelines. When a decision is made to provide an item or an action specified in a guideline, the utility should address a set of questions relating to the design, operation and availability of the needed equipment and the training of operators. For the example of containment-spray guidance, the capacity of the spray system, the selection of setpoints to initiate sprays, the availability of applicable procedures and the accessibility of certain valves by operators should be addressed. The criteria on containment spray provide helpful information in assessing spray capability in each individual plant.

The criteria address the general issues of (1) survivability of equipment (i.e., whenever credit is given for a system or a component to mitigate the accident, the ability of the equipment to function under environmental and fluid dynamic loads associated with severe-accident sequences must be taken into account), (2) equipment capabilities, capacities, and duration of operability, (3) accessibility of equipment, (4) availability of support systems,

(5) identification of necessary components, (6) identification of important operator actions, and (7) identification of parameters for initiation of mitigating systems and operator actions.

2.3 Organization of the Report

This report describes detailed guidelines and criteria for preventing and mitigating severe accidents in PWRs that have an ice-condenser containment. It is the last volume in a series of five, that deal with guidelines and criteria for several different reactor and containment types. Other volumes in the series are:

- Volume 1: BWRs with Mark I Containments
- Volume 2: BWRs with Mark II Containments
- Volume 3: BWRs with Mark III Containments
- Volume 4: PWRs with Large-Volume Containments.

Appendix A of this volume contains a review of the IDCOR and SARP analyses for a PWR with an ice-condenser containment along with other pertinent studies. The insights gained from these studies lead to the identification of the strengths and vulnerabilities of a PWR with an ice-condenser containment. In Section 3, the three basic goals of the program are related to the relevant design features and operating characteristics of a PWR with an ice-condenser containment. The guidelines recommended to achieve the three goals are therefore initially developed in Section 3. In Section 4, the guidelines are restated and detailed criteria are developed for each guideline.

2.4 References for Section 2

1. "Sequoyah Nuclear Power Plant - Integrated Containment Analyses," IDCOR Technical Report T23.1S, March 1985.

2. R. C. Bertucio et al., "Analysis of Core Damage Frequency from Internal Events: Sequoyah, Unit 1," Sandia National Laboratories, NUREG/CR-4550, Draft, Volume 2, February 1987.

3. V. L. Behr et al., "Containment Event Analysis for Postulated Severe Accidents: Sequoyah Power Station, Unit 1," Sandia National Laboratories, NUREG/CR-4700, Volume 2, Draft Report for Comment, February 1987.

4. A. S. Benjamin et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Sequoyah Power Station, Unit 2," Sandia National Laboratories, NUREG/CR-4551, Volume 2, Draft for Comment, February 1987.

5. "Reactor Risk Reference Document," U.S. Nuclear Regulatory Commission, NUREG-1150, Draft for Comment, February 1987.

3. DEFINITION OF GOALS AND RELEVANT FEATURES OF PWRS WITH ICE-CONDENSER CONTAINMENTS

In Section 2 of this report the concept of three basic objectives or goals for this severe ident program was introduced. The concept applies equally to all plant types. In this section, the three goals are related to the relevant design features and operating characteristics of a PWR with an ice-condenser containment for the accident sequences and containment failure modes found to be important in Appendix A. This includes consideration of both favorable and unfavorable severe accident attributes. Table 3.1 summarizes the dominant accident sequences identified in the Sequoyah ASEP study. PRAs for other PWRs have been reviewed for the possibility of unique sequences which may contribute to CDF for other ice-condenser plants.

Screening criteria have been used to identify those sequences which need to be addressed by severe accident guidelines for each goal. Specifically:

- For Goal 1 (Mitigate fission-product releases), all sequences have been examined which represent 5% of the core-melt frequency or are estimated to occur more often than 10^{-6} per reactor-year and which result in a conditional probability of early containment failure greater than 0.1.
- For Goal 2 (Control the frequency of high-consequence sequences) all sequences have been examined which result in containment bypass and are estimated to occur more often that 10^{-7} per reactor-year.
- For Goal 3 (Reduce high core-damage frequency sequences) all sequences have been examined which "have the potential to occur" more frequently than 10^{-6} per reactor-year. Note that this screening criterion has been used to identify potential vulnerabilities from risk assessment insights which do not necessarily apply to Sequoyah itself, but may apply to other PWR ice-condenser plants.

This section provides the link between the goals (developed in Section 2) and the guidelines (developed in Section 4), that will be used to assess the capability of specific plants to meet these goals. This section is organized into three subsections, which correspond to the three goals.

3.1 Mitigate Fission-Product Releases

This goal requires that there shall be effective means of mitigating the fission-product releases for the broad classes of accident sequences which may lead to core damage in an ice-condenser plant. In Appendix A, the most important contributors to the core-damage frequency (CDF) were found to be small breaks (including the interfacing systems LOCAs and reactor coolant pump seal LOCAs), large breaks, and transients with feedwater failure (including station blackout). Specific accident sequences for which mitigation by the ice-condenser containment is ineffective were also identified in Appendix A and they are discussed in Section 3.2. This section concentrates on the broad classes of accident sequences for which plant features provide significant means of mitigating fission-product release. In the following subsections both the favorable and unfavorable severe accident attributes of the ice-condenser containment will be identified. This discussion, in turn, leads to the development of three guidelines related to Goal 1.

3.1.1 Plant Vulnerabilities

As noted in Appendix A, the Sequoyah containment is a medium-volume pressure suppression design. The ice condenser (see Figure A.1), is the primary pressure-suppression component. Inlet and outlet doors are provided at the bottom and top of the ice-condenser compartment. In the event of a LOCA, the lower inlet doors will open because of the pressure rise in the lower compartment caused by the release of the reactor coolant to the lower compartment. The differential pressure will then cause air, entrained water, and steam to flow from the lower compartment into the ice condenser. An operating deck separates the upper and lower compartments and ensures that steam and air flow resulting from a LOCA is directed through the ice condenser to the upper compartment rather than through uncontrolled bypass paths. There are two pathways for water from melted ice and containment sprays to flood the reactor cavity from the lower compartment. The first pathway is through the reactor vessel nozzle penetrations in the reactor shield wall. The second pathway is for water to accumulate above the personnel access hatch flooding the cavity via the instrument tunnel.

The limited volume of the ice-condenser containment makes it vulnerable to pressure buildup because of noncondensable gas generation and hydrogen combustion during a core-melt accident. There are differences between the IDCOR[1,2] and ASEP/SARRP[3-5] analyses as to how long it will take to pressurize an ice-condenser containment to its ultimate capacity after the core debris has failed the reactor vessel (and it is interacting with concrete) but both studies concluded that for some cases containment failure will eventually occur. Therefore, unless mitigative actions are taken, an ice-condenser containment will fail for some accidents because of overpressure within 10 to 30 hours of vessel failure. If containment failure occurs, a significant inventory of fission products in the containment atmosphere could be released to the environment.

An inspection of the ice-condenser containment configuration in Figure A.4 indicates that the cavity below the reactor pressure vessel would tend to confine the core debris after a core-melt accident at low pressure. Under normal conditions the water from continuous ice melting and containment sprays will fill the upper reactor cavity and overflow into the lower cavity, cooling the core debris. In the case of blocked drains, the containment spray will accumulate in the upper compartment and the melted ice may be insufficient to overflow into the lower cavity at the time of vessel failure. Thus, the cavity may remain dry. In the absence of a water supply, extensive core-concrete interactions would be expected to occur. There are differences between the IDCOR[2] and BCL[5] analyses as to how hot the core debris will be during core-concrete interactions and as to how much of the less volatile fission products will be released. However, at this time the possibility of the core debris remaining hot or eventually drying out and releasing significant quantities of fission products has not been ruled out.

The most important vulnerability of the ice-condenser containment appears to be the possibility of early containment failure, because of hydrogen deflagrations or detonations or direct heating. SARRP predicts that most of the early containment failures are because of hydrogen combustion. Otherwise containment failure (if it occurs) is expected to be late overpressure failure because of buildup of noncondensable gases or steam after depletion of the

ice. Therefore, the probability of hydrogen detonations and deflagrations should be controlled. The probability of detonation can be controlled by ensuring the proper function of the igniters. However, deflagrations may still occur whether or not the igniters function. The key to eliminating deflagration appears to be RCS depressurization to allow controlled hydrogen burns. The atmosphere recirculation from the containment back into the ice bed is also beneficial. It provides a uniform mixture of air and hydrogen, cools the atmosphere and facilitates removal of fission products from the containment air.

In addition to the hydrogen deflagrations and detonations, the interfacing systems LOCA has the potential to release substantial fission products without the benefit of scrubbing by the ice and/or containment sprays.

3.1.2 Mitigating Features

The ice condenser is an effective device for removing any fission product aerosols that might pass through it. Thus, to some extent the ice condenser has the potential to compensate for the vulnerabilities identified above (in Section 3.1.1).

Overpressure failure of the containment can be prevented by intensive cooling of ECC water and cooling of recirculating spray water. The IDCOR calculations[2] indicated (see Table A.16) that about twice the heat removal rate assumed by BCL[5] (i.e., two pumps) from the recirculating spray water could prevent overpressure failure of the containment.

The most severe consequences of core-melt accidents in Sequoyah are from accidents with early containment failure. The SARRP uncertainty analysis[4] indicates that hydrogen detonations or deflagrations may cause such early failure. In order to prevent hydrogen detonations and deflagrations, the hydrogen igniters must be operable and large hydrogen concentration gradients must be prevented.

3.1.3 Maintain Containment Integrity

The above discussion has identified several features of the ice-condenser plant that have the potential to help achieve Goal 1, namely, mitigating fission-product releases. Therefore, several guidelines have been developed and related to those features which will aid in assessing whether specific plants can meet Goal 1. The guidelines address the following areas:

- containment sprays and long-term heat removal
- filtered venting, and
- reliable operation of ICS and ARFS

3.2 Control the High-Consequence Sequences

The plant features identified in Section 3.1 can effectively mitigate fission-product releases for the broad classes of accident sequences that were found to dominate the CDF (see Table 3.1). However, accident sequences were found in Section A.1 for which mitigating the fission-product release cannot be assured.

3.2.1 Interfacing Systems LOCA (Including Steam Generator Tube Rupture (SGTR))

The interfacing systems LOCA (ISL) would open up a direct path from the primary system to the reactor auxiliary building which bypasses the ice-condenser and the primary containment. The only plant feature pertinent to mitigating this event is then the reactor auxiliary building, which may not be sufficient on its own to ensure low fission-product releases to the environment. Although Appendix A indicates that the ISL is not a significant contributor to core damage for Sequoyah, PRAs for other PWRs have identified it as a significant contributor to risk. It has also been identified as a generic issue by the NRC (GI-105). In Section 4.2, a guideline and associated criteria are developed to try to ensure that other ice-condenser plants assess its contribution to risk. It is noted that Brookhaven National Laboratory (BNL) is currently performing a study on interfacing systems LOCA to provide technical support to the NRC for a meaningful resolution of this generic issue.

The steam generator tube rupture (SGTR) event would provide an open pathway for early radioactive release to the atmosphere if the core damage is not arrested in a timely fashion and the release paths in the secondary side, e.g., the main steam isolation valve (MSIV) of the ruptured steam generator (SG), and the associated SG atmospheric steam dump valves are not isolated properly. Although this initiator has not been identified as a dominating contributor to the overall CDF for Sequoyah, it has been identified as a contributor to risk for other PWRs. Thus, the frequencies of these sequences should also be controlled (Goal 2). In Section 4.2, a guideline and associated criteria are developed to ensure that other PWRs with ice-condenser containments do not have a high risk because of containment bypass events.

3.3 Reduce High Core-Damage Frequency Sequences

The set of accident sequences which were identified in the IDCOR[2] and ASEP[3] studies as dominant contributors to CDF for the Sequoyah plant are presented in Section A.1. Although the number of contributing sequences is large, a relatively few number of sequences (8 sequences are greater than 10^{-6}/reactor year in Table 3.1) are key contributors to CDF. The most important contributors to the CDF were found to be related to small LOCA (S_2), station blackout, and loss-of-component cooling water (T_{CCW}) sequences.

3.3.1 Operator Response for Recirculation

Major contributors to the LOCA sequences are related to the high-pressure recirculation switchover which involves several operator failures and the common cause failures of the containment sump suction valves and high-pressure injection pump suction valves. This has been found also in other PWR PRAs even though emergency operating procedures of the plants usually address this recirculation switchover operation.

3.3.2 Station Blackout

Characteristics of the important accident sequences in station blackout (SBO) are dc bus failure or battery depletion defeating the turbine-driven AFWS pump, and RCP seal LOCA because of failures of seal injection flow and

failures of RCP thermal barriers resulting from loss of electric power to the component cooling water and service water systems. SBO is currently the subject of an unresolved safety issue (USI A-44) while RCP seal failure is the subject of a generic issue (GI-23). Thus, development of these related guidelines and criteria are considered to be appropriate pending resolution of USI A-44 and GI-23.

3.3.3 Component Cooling Water (CCW)

Loss-of-component cooling water (CCW), as an initiator, may cause failures of seal injection flow and failure of the thermal barrier and can lead to reactor coolant pump (RCP) seal LOCA sequences because of loss of RCP seal cooling. Note also, that, depending on the redundancy and heat loads of the CCW system and the essential service water (ESW) system in a particular plant, the loss of ESW could be more "critical" than the loss of CCW with regard to RCP seal LOCA, depending on whether the charging pumps (which provide RCP seal injection water) are cooled by the CCW or by the ESW system. Even for RCP seals which are cooled by CCW, an RCP seal LOCA may result from the loss of the ESW since the loss of ESW, if it is not rapidly recovered would lead to degradation of CCW. In Section 4.3, guidelines and associated criteria are developed for the CCW to ensure that each plant assess the likelihood of CCW failures leading to core-melt.

3.3.4 Reactor Coolant System (RCS) Depressurization

Small break LOCAs with failure of the high-pressure injection system (HPIS) have been found to be a contributor to CDF for many PWR risk assessments. Changes to emergency procedures to allow RCS depressurization by secondary blowdown are being reviewed by the nuclear industry and the NRC. Although the net risk reduction has yet to be assessed for ice-condenser plants, it is considered prudent to provide a guideline to assess this capability at each plant.

3.3.5 Reactor Coolant System (RCS) Feed and Bleed Cooling

Table A.6 indicates that failures of the auxiliary feedwater system (AFWS) and the feed and bleed cooling operation are significant contributors to the core damage in SBO or loss of dc bus sequences. The success of RCS feed and bleed cooling (high pressure injection (HPI) and steam relief through the RCS power operated relief valves (PORVs)) has a significant impact on the estimated CDF for many PWRs. However, the capacity of the PORVs and the HPI varies from plant to plant. Thus, a guideline is provided to assess RCS feed and bleed capacity and to ensure that the operator training and emergency procedures are sufficient to achieve successful feed and bleed cooling if it is feasible. Note that feed and bleed cooling is related to the decay heat removal issue (USI A-45).

3.3.6 Anticipated Transients Without Scram (ATWS)

Another set of accident sequences that may defeat several plant safety features and that contribute to the CDF is related to an ATWS. ATWS is a small contributor (1.1×10^{-6}) to CDF for Sequoyah. It is also a dominant contributor for other PWRs including Surry and Millstone-3. The ATWS initiated by loss of main feedwater was found to be most severe. Failure of secondary

cooling via the SG and RCS PORV relief in this sequence could lead to a small LOCA and damage the check valves between the RCS system and the HPI systems which may lead to core damage. In particular, if the turbine fails to trip in this sequence, its demand for steam on the SGs will cause them to dry out rapidly, rendering the SG cooling ineffective.

3.3.7 Support System Interdependencies

Most PRAs have stressed the importance of unrecognized interdependencies having the potential to compromise the performance of many critical safety systems. In many cases risk assessment studies have identified such vulnerabilities very early in the study and fixes have been made which substantially reduced risk.

The ASEP study of Sequoyah indicated a vulnerability to loss of CCW. PRAs for other PWRs have also found vulnerabilities to service water and ac and dc busses. Thus, it is extremely important that each plant attempt to identify any unique support system vulnerabilities.

3.3.8 Flooding of Emergency Equipment

As excessive water release into a portion of the turbine building or auxiliary building outside the primary containment which houses a concentration of safety-related equipment raises the possibility of a common mode event disabling key equipment. At least one PWR has been identified i.. which the location of safety equipment makes the internal flooding initiator a substantial contributor to CDF.

To help ensure that other PWR plants which may have a similar safety-related equipment flooding potential can be identified, a guideline and associated criteria are provided in Section 4.3 which may be used to screen for such vulnerabilities.

3.4 References for Section 3

1. "Risk Reduction Potential," Energy Incorporated, IDCOR Technical Report 21.1, June 1985.

2. "Sequoyah Nuclear Power Station - Integrated Containment Analysis," IDCOR Technical Report T23.1S, March 1985.

3. R. C. Bertucio et al., "Analysis of Core Damage Frequency from Internal Events: Sequoyah, Unit 1," Sandia National Laboratories, NUREG/CR-4550, Draft, Volume 2, February 1987.

4. A. S. Benjamin et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Sequoyah Power Station, Unit 1," Sandia National Laboratories, NUREG/CR-4551, Volume 2, Draft for Comment, February 1987.

5. R. S. Denning et al., "Report on Radionuclide Release Calculations for Selected Severe Accident Scenarios to U.S. Nuclear Regulatory Commission," Battelle Columbus Laboratories, NUREG/CR-4624, BMI-2139, Vol. 5, July 1986.

Table 3.1 Dominant Accident Sequences for Sequoyah in the ASEP Study (Point Mean Values Per Reactor Year)[a]

Sequences[b]	Core-Damage Frequency
V	3.3E-7
S_2H_2	2.9E-5
T_{CCW}	2.7E-5
S_2H_3	9.7E-6
$T_1L_1D_1F$	8.3E-7
$T_1D_3WD_1F$	2.9E-6
S_2H_3F	7.7E-6
$T_1Q_1-D_1F$	3.3E-7
S_1H_2	1.6E-6
$T_{DCI}L_1P_1$	1.1E-6
$T_{DCII}L_1P_1$	1.1E-6
TOTAL	8.2E-5 (1.0E-4)[c]

[a]From information contained in the ASEP report.[3]
[b]See the glossary for Table A.6 in Appendix A.
[c]Mean core-damage frequency obtained from Monte Carlo simulation.

4. GUIDELINES AND CRITERIA FOR A PWR WITH AN ICE-CONDENSER CONTAINMENT

In Section 3, those accident sequences that dominate the core-damage frequency (CDF) were identified and listed in Table 3.1. The interfacing systems LOCA (V sequence) was identified as having potentially high consequences for ice-condenser plants. Vulnerabilities of the ice-condenser containment to severe accident containment loads were discussed, and those features of a PWR with an ice-condenser containment which are important for preventing core damage and are available for mitigating fission-product releases to the environment were identified.

Based on the insights from the IDCOR and ASEP/SARRP studies for Sequoyah and previous PRA studies, the following sections provide guidelines defining "deterministic, plant-specific guidance on the design features and operating characteristics which are to be examined by the utilities,"[1] and criteria defining "deterministic standards for judging the acceptability of plant features."[1] From SECY-86-76[2] further guidance is provided in defining guidelines and criteria. These guidelines "will specify the plant features and operator actions which are considered important to ensuring acceptable risk for the reference plant."[2] Further acceptance criteria (for the various guidelines) "will specify the attributes necessary to ensure acceptable performance."[2]

Based on this work, twelve guidelines were developed which reflect the importance of these features to plant risk. As discussed in Section 2.2.1 these guidelines indicate areas of potential improvements for various areas of plant design and operation of which utilities should be aware when conducting assessments. It is further noted that a number of the guidelines appear to overlap various generic issues as defined by the NRC. Final resolution and disposition of these generic issues may encompass NRC-imposed requirements. However, the guidelines and criteria presented herein are intended only for the purposes noted above. The guidelines are summarized in Table 1.1.

Guidelines 1 through 3 were developed to ensure the capability to mitigate fission-product releases (Goal 1) with reference to maintaining containment integrity, maintaining ice-condenser system (ICS) effectiveness and providing reliable hydrogen control.

Guideline 4 was developed for controlling the frequency of high-consequence sequences (Goal 2) with reference to minimizing interfacing systems LOCA frequency.

Guidelines 5 through 12 were developed for reducing high CDF sequences (Goal 3) with reference to minimizing recirculation failure, mitigating station blackout (SBO) sequences, mitigating loss of component cooling water (CCW) sequences, enhancing reactor pressure vessel (RPV) depressurization capability, examining feed and bleed cooling capability, mitigating anticipated transients without scram (ATWS) sequences, examining support system interdependencies, and mitigating internal floods.

The remainder of this section is organized into three subsections corresponding to the three basic goals. In each subsection, the corresponding guidelines are discussed from which detailed criteria are developed in order to provide the standards by which each plant will be measured for compliance with the guidelines. The criteria address (see Section 2.2.2), under severe

accident conditions, the general issues of (1) survivability of equipment (i.e., the ability of the equipment to function under the environmental conditions and fluid loads associated with severe accident sequences), (2) equipment capabilities and duration of operability, (3) accessibility of equipment, (4) availability of support systems, (5) identification of necessary components, (6) identification of important operator actions, training and procedures, and (7) identification of parameters for initiation of mitigating systems and operator actions.

4.1 Mitigate Fission-Product Releases

For a PWR with an ice-condenser containment, the dominant core-damage sequences were found to be small break LOCAs (including pump seal LOCAs and ISL) and transients (including SBO). In order to minimize off-site consequences, the containment systems must be able to fulfill their role of restricting fission-product releases even under these severe accident conditions. In this section, three guidelines are provided with associated criteria which should help other PWRs assess the capability of their containments to mitigate fission-product releases during a severe accident.

4.1.1 Maintain Containment Integrity (Guideline 1)

For sequences that result in corium ejection from the reactor pressure vessel (RPV) at high pressure, water injected via the containment sprays combined with melted ice can overflow into the reactor cavity with the potential to help cool the debris, thus preventing early containment failure by containment shell melt-through or overpressurization because of direct heating. For many of the dominant sequences SARRP predicts long-term overpressure failure of the containment near the time of vigorous core-concrete interaction with a concomitant substantial fission-product vaporization release. A guideline to provide an assessment of the containment capability to withstand early threats to containment has been suggested for PWRs with large-volume containment. However, the ice-condenser containments are very similar to each other in design and any threats to the Sequoyah containment appear to apply generically to the other plants as well. Therefore, no additional guidance on assessment of containment performance is needed. The sprays provide substantial containment heat removal capability and will delay or prevent containment failure. In some sequences, such as station blackout, the sprays may not be available because of ac power requirements while in other sequences may deplete the refueling water storage tank (RWST) inventory and the sprays may fail in the recirculation mode.

Table 4.1 provides the criteria developed for this guideline to evaluate the effectiveness of containment sprays and filtered venting systems in maintaining containment integrity.

4.1.2 Ice-Condenser System (ICS) and Air Return Fan System (ARFS) (Guideline 2)

In the Sequoyah plant, the ice condenser plays an essential role in condensing steam and scrubbing fission products for a broad range of accidents. In order to preserve these functions, it is desirable that the ARFS remain operable and ice-condenser bypass conditions be avoided. The ARFS helps the ICS and the containment spray system by circulating air between the lower and

upper containment compartments through the ice condenser. The ARFS also promotes mixing of hydrogen with air in the containment. These two design features for Sequoyah permitted a lower containment design pressure and a smaller containment volume than those for a PWR without an ICS (e.g., large-volume containment). The unavailabilities of these two systems in the event of an accident would be important to risk since the containment could rapidly pressurize and fail without sufficient heat removal capability in the ice chest.

Table 4.2 provides the criteria developed for this guideline to evaluate the effectiveness of these systems under the conditions imposed by the dominant severe accident sequences.

4.1.3 Hydrogen Control (Guideline 3)

The most severe consequences of core-melt accidents in Sequoyah are from accidents with early containment failure. The SARRP analyses indicate that large deflagrations and hydrogen detonations may cause such early failure. The hydrogen ignition system promotes combustion of the hydrogen at low concentrations in the containment air, thereby avoiding the large pressure spikes associated with hydrogen detonation or deflagration and preserving containment integrity. In addition to the igniters themselves, the placement of the igniters and the availability of the ARFS are important to avoid locally high concentrations of hydrogen. The present igniter system can operate only if ac power is available.

Table 4.3 provides the criteria developed for this guideline to evaluate the effectiveness of each plant's hydrogen ignition system in mitigating the dominant severe accident sequences.

4.2 Control the Frequency of High-Consequence Sequences

Accident sequences were found in Section A.1 for which the PWR with an ice-condenser containment has limited means of mitigating fission product releases. In this section, guidelines and associated criteria for controlling these potentially high-consequence sequences are developed.

4.2.1 Interfacing Systems LOCA (Including Steam Generator Tube Rupture) (Guideline 4)

As discussed in Section 3.2, although the interfacing systems LOCA (ISL) has been found to be a highly unlikely event, it can be a significant risk contributor for some plants because of the potentially high fission-product releases to the environment. The objective of this guideline and associated criteria is to ensure that the frequency of ISL events is kept at an acceptably low level.

Brookhaven National Laboratory (BNL) is presently performing a study to provide technical support to the NRC for the meaningful resolution of the generic issue related to ISL (GI-105). For the ISL guideline, the performance of equipment, systems, and operators should be assessed against specific performance criteria to ensure successful prevention of interfacing systems LOCA. The criteria relate to equipment (low-pressure systems interfacing with high-pressure systems) and operator performance (isolation and relief valve maintenance and surveillance).

Although steam generator tube rupture (SGTR) bears the characteristic of a small LOCA, it is unique in the sense that it is also a potential containment bypass LOCA, releasing primary reactor coolant into the secondary side of the steam generators (SG), which provides several potential paths for release of fission product to the environment outside the containment via the main steamline, turbine, turbine bypass, etc. Thus, despite its small contribution to risk,[3] it is important to provide the necessary measures to control the likelihood of occurrence of SGTR and to isolate an SGTR if it occurs.

Prevention of SGTR frequency should be aimed at removing all significant causes that may contribute to degradation and failure of SG tubes, with special emphasis placed on preventing chemical corrosion of the tubes by improved secondary side water chemistry, and/or protecting the integrity of steam generator tubes from the impact of foreign objects through improved surveillance and inspection. The primary objectives of recovery actions and mitigating the consequences of SGTR should be (1) to stop the primary-to-secondary leakage, (2) to restore reactor coolant system (RCS) inventory to ensure adequate core cooling, (3) to minimize the release of fission products to the surrounding environment from the ruptured SG, and (4) to stabilize the reactor system to regain plant control.

Several important operator actions are required for successful achievement of these objectives, including early and correct diagnosis of SGTR, identification as well as isolation of the faulty SG, manual depressurization of the RCS to stop the leakage flow and prevention of main steam line flooding. Since some of the key operator actions differ substantially depending upon whether the offsite power is available or not, the operators should be trained to familiarize themselves with the correct measures that must be taken under both conditions.

The criteria developed for the guideline are presented in Table 4.4. These criteria should be considered as appropriate pending final resolution of the generic issue on ISL (GI-105) and the unresolved safety issues on SGTR (USIs A-3, A-4 and A-5).

4.3 Reduce High Core-Damage Frequency Sequences

The major contributors to core-damage frequency (CDF) for Sequoyah are identified in Appendix A and include small LOCA (S_2), station blackout (SBO), and loss of component cooling water (T_{CCW}) sequences. Thus, if the frequencies of these accident sequences (or subset of them) can be reduced, then the overall CDF will be substantially reduced. In this section, guidelines and criteria for reducing high CDF sequences are developed.

4.3.1 Operator Response for Recirculation (Guideline 5)

As presented in Section A.1, the small LOCA initiator (S_2) has been found to be the largest contributor to CDF at Sequoyah. Major contributors to the S_2 sequences are operator failures in the recirculation switchover operation and common cause failures of the containment sump suction valves and the high pressure injection pump suction valves. The recirculation switchover operation requires several operator actions to realign flow paths after the low refueling water storage tank (RWST) alarm is given:

i) changing positions of several valves downstream of the low-pressure injection pumps,
ii) isolation of the emergency core cooling system (ECCS) from the RWST suction,
iii) switchover of containment spray system (CSS) to recirculation, including valving in service water (SW) to RHR heat exchangers, and
iv) valving in component cooling water (CCW) to the RHR heat exchangers.

Although the sequences involving loss of containment heat removal (CHR) were found only marginally important contributors to core-damage sequences for Sequoyah (S_1F, S_1G_1, AF, AG_1 in Table A.6), a main contributor to such sequences in other PRAs is often related to operator error in the recirculation switchover operation. Since this switchover operation has also been found important in other PRAs, the individual plant's emergency operating procedures for many plants have been changed to include detailed descriptions of the steps involved in the switchover operation. It is suggested in Section A.1.4 that the operator failure probabilities will be minimized if the operators are trained in such a way that these several actions during switchover are an integral operation for a single function. Thus, the likelihood of human error could be substantially reduced.

The criteria developed for this guideline in Table 4.5 are based on this consideration and on eliminating common cause failures.

4.3.2 Station Blackout (Guideline 6)

In most PRAs for light-water-reactors (LWRs), station blackout (SBO) sequences have been major or prominant contributors to the CDF. The accident sequences in SBO are usually characterized by two categories of events: the failure to recover ac power before battery depletion (defeating the turbine-driven auxiliary feedwater system (AFWS) pump) and the RCP seal LOCA (because of failures of seal injection flow and RCP thermal barriers resulting from loss of the CCW and service water systems).

As part of the effort to resolve the unresolved safety issue (USI A-44), the NRC is proposing to amend its regulations "to provide further assurance that a station blackout (loss of both offsite power and onsite emergency ac power systems) will not adversely affect the public health and safety."[7] For accident sequences, developed by an individual plant examination (IPE), which involve the loss-of-offsite power and onsite emergency power, the proposed SBO rule should be examined for applicability. The criteria associated with Guideline 6 are intended to emphasize the need to search for plant specific features and potential common cause failures which could disable systems required to work during an SBO. For individual plants which are found to have a vulnerability to SBO, the criteria given in Table 4.6 highlight the importance of proper emergency procedures and operator training in recovering from an SBO event.

4.3.3 Component Cooling Water (CCW) (Guideline 7)

An event initiated by loss of CCW may lead to RCP seal LOCA sequences because of loss of RCP seal cooling, under the condition that high-pressure injection (HPI) also fails when these pumps are also cooled by CCW. In some plants the RCP seal injection is accomplished by the essential service

water (ESW). RCP seal LOCA may also result from loss of the ESW since its loss, if it is not recovered in time, would lead to failure of the CCW system. Thus the criteria modifications may also be cost beneficial. The CDF because of seal LOCA can also be reduced by making the RCP seal cooling less susceptible to failure (e.g., using alternate seal injection systems[5] or steam-driven and self-cooled charging pumps[5]) or by improving the seal designs (e.g., using a pneumatic seal which prevents leakage[7]). The criteria developed for this guideline are presented in Table 4.7 and are considered appropriate pending the resolution of the generic issue on RCP seal failure (GI-23).

4.3.4 Reactor Coolant System (RCS) Depressurization by Secondary Blowdown (Guideline 8)

One potentially effective means of making up core inventory in case the HPI systems are unavailable (either in injection or in recirculation) is to depressurize the reactor coolant system (RCS) by heat removal through the steam generators (SG). After the reactor coolant system has been depressurized via secondary heat removal, the low-pressure injection systems (LPIS) can be actuated for RCS makeup operation. This emergency procedure, currently being implemented at Sequoyah based on Westinghouse Owners Group Emergency Response Guidelines, may have significant impact on reducing the CDFs for S_1H, S_2H, S_1D, and S_2D sequences, as pointed out in Section A.1.1.3. However, a decay heat removal study performed by Sandia National Laboratories[13] (SNL) indicates that credit for this procedure has very little impact on CDF for a Combustion Engineering plant. Feed and bleed capability (Guideline 9) was found to be much more beneficial in reducing overall CDF (by about a factor of 3). The impact of RCS depressurization on S_1H and S_2H sequences should be significant because this procedure makes the dominant contributor (operator failure in performing the high pressure recirculation switchover) no longer predominant. To attain success in this procedure, the operator must open the SG atmospheric steam dump valves, maintain AFW or main feedwater (MFW) to the SG and have one of the LPIS trains available. Improved depressurization capability also has the potential to reduce the likelihood of early containment failure by reducing the magnitude of the pressure spike because of direct heating and hydrogen combustion at the time of vessel failure. Although the net risk reduction for these low-pressure injection (LPI) procedures has not yet been assessed by the NRC, the potential looks promising and criteria have been developed pending further analysis.

The criteria developed for this guideline are presented in Table 4.8.

4.3.5 Reactor Coolant System (RCS) Feed & Bleed Cooling (Guideline 9)

The success of RCS feed & bleed cooling (high-pressure injection (HPI) and steam relief through the RCS power operated relief valves (PORVs)) has a significant impact on reducing the CDF associated with sequences involving auxiliary feedwater system (AFWS) failures. Successful RCS feed & bleed cooling hinges on proper operator actions as well as on operability of the HPI systems and the PORVs, and the PORV block valves. The operator must ensure sufficient injection by charging pumps or HPI pumps and manually open the required number of PORVs. Appropriate containment heat removal systems must also be available. The use of RCS feed and bleed as a backup to the AFWS is being addressed as part of the decay heat removal issue (USI A-45).[13]

The criteria for this guideline should be considered as appropriate pending resolution of USI A-45 and are given in Table 4.9.

4.3.6 Anticipated Transients Without Scram (ATWS) (Guideline 10)

The guideline for ATWS sequences assumes compliance with the NRC rule on "reduction of risk from ATWS events for light-water-cooled nuclear power plants."[8] Also, in the NRC rule on ATWS, "the Commission has concluded that a reduction in the frequency of challenges to plant safety systems should be a prime goal of each licensee, and the Commission believes that ATWS risk reduction can also be achieved by reducing the much larger frequency of transients which call for the reactor protection system to operate."[8] As indicated in Table A.7, ATWS is not a dominant contributor to CDF for Sequoyah and this is also generally supported by PRAs for other Westinghouse plants. One factor in these results is the credit given to the manual scram and emergency boration using the charging system to deliver borated water to the reactor pressure vessel. Failures of manual scram and emergency boration are dominated by operator action failures. These operator actions are not specifically addressed in the ATWS rule.

Detailed criteria developed for this guideline are given in Table 4.10 and are based upon the assumption that each of the plants is (or will be) in compliance with the NRC rule on "Reduction of Risk from Anticipated Transients Without Scram for Light-Water-Cooled Nuclear Power Plants."[8]

4.3.7 Support System Interdependencies (Guideline 11)

One of the primary benefits of performing a rigorous PRA is that the system interdependencies are modeled and are reflected in the results. However, not all PRA studies have performed in-depth interdependence analyses and therefore have not ferreted out all of the possible subtle interdependencies. This may have profound effects upon their results. A dependency is defined as the failure of one system leading directly or indirectly to the failure of another system.

An in-depth application of basic PRA methodology with respect to interdependencies yielded significant findings on a previously heavily studied PWR. To illustrate this point, reference is made to the BNL study of system interactions (support system interdependencies) at Indian Point Unit 3.[9] The major finding of that study was that a specific single station emergency battery could fail and among other things, negate the entire low pressure injection function. The point to be emphasized here is that none of the numerous other studies and reviews of the Indian Point 3 design were able to detect this important single failure nor did the BNL study until all the support systems were explicitly modeled, linked together (the fault tree linking approach[10]) and solved using the SETS computer code.[11]

NUREG-1150[12] has provided a thorough application of the latest PRA methods to five reference plants and the results point out numerous insights into the importance of specific design differences among the studied plants. However, the NUREG-1150 authors emphasize the importance of support system differences and the difficulty of extrapolating the result of one plant to another.

It is not sufficient to make a single overall dependence table of the front-line and support systems for a given plant and simply compare that to the reference plant. No two plants will have the same set of system interdependencies. It is known that in U.S. nuclear plants, support systems vary widely from plant to plant even though the plants may be of a similar class and have the same set of front-line systems.

It is recognized that following the interdependency evaluation steps outlined in Table 4.11, in a rigorous fashion, is a major undertaking. This fact, however, does not diminish its importance.

Based upon the dominance of the SBO sequence and the loss of CCW sequence for Sequoyah as well as other PWR designs, it is recommended that detailed interdependency tables be constructed for the SBO and loss of CCW sequences with all dependencies conditioned upon the existence of a SBO for various lengths of time and upon the loss of CCW, respectively. These tables should also explicitly identify all of the expected failure mechanisms (e.g., identify whether batter failure is because of loss of room cooling or charge depletion).

4.3.8 Flooding of Emergency Equipment (Guideline 12)

Although medium or small leakages can be adequately mitigated by the existing sumps or pumpback systems, large water leakages are of primary concern in flooding. Potential water sources for excessive water release into the turbine building or auxiliary building include the condensate storage tank, the reactor coolant system, the service water system, the refueling water storage tank, and the fire protection system storage tank. Some of the major equipment located in the auxiliary building may include emergency core cooling (ECC) pumps and their electrical control panels.

Flooding can be initiated by (1) a major maintenance and (2) breaks in the pressurized or non-pressurized part of piping or components. In item 1, "major maintenance" refers to those actions which would require dismantling of system components thus eliminating a barrier between large sources of water and the auxiliary building. Flooding can partly be prevented and/or mitigated through proper training and procedures. For example, once the flooding occurs, the operator should be able to follow the instructions for responding to the alarm to identify the source of the flood and isolate it before the water level in the compartment reaches a critical level. The operator should also know about alternative devices or equipment which can be utilized to provide coolant injection to the RPV in case the emergency core cooling (ECCS) systems equipment fail in the flood compartment.

Although this type of vulnerability to flooding was identified for only one PWR, it is believed that the concerns are of general applicability to other designs. Thus, it is recommended that all PWR plants be screened for flooding vulnerabilities.

Detailed criteria developed for this guideline are given in Table 4.12.

4.4 Using the Guidelines and Criteria

Numerous investigations, including PRAs, have been performed for the reference plants and for similar plants by both the NRC and the nuclear power industry. The insights gained from many of the studies have been used in developing the guidelines and criteria contained in this report (including the other volumes relating to other plant types). The guidelines and criteria are issued to guide the analyst performing an IPE. This guidance is in the form of plant features, operator actions and the criteria for assessing those features and actions found to be helpful in reducing the overall risk for Sequoyah and other PWR plants. Thus, the guidance is given to provide a resource in examining the subject plant to determine if the same, or similar, guidelines will be of value in reducing overall plant risk. These guidelines and criteria are intended to be used solely as guidance, but they may include (as a subset) some requirements generated by the NRC on generic issues.

4.5 References for Section 4

1. R. Barrett, "Status of the Severe Accident Program for Operating Reactors," NRR Staff Presentation to the ACRS Subcommittee Class 9 Accidents, February 24, 1986.

2. SECY-86-76, "Implementation Plan for the Severe Accident Policy Statement and the Regulatory Use of New Source-Term Information," NRC/EDO, February 28, 1986.

3. "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," U.S. Nuclear Regulatory Commission, NUREG-0844, Draft Report for Comment, April 1985.

4. NRC Station Blackout Proposed Rule, Federal Register, Volume 51, No. 55/March 21, 1986, pgs. 9829-9835.

5. E. Silvestri et al., "A Model for the Probability of Core Uncovery in LOOSP Induced Accidents, as Applied in the Probabilistic Safety Study for ENEL PWR Standard Power Plant," Paper No. 109, Proceedings of the International ANS/ENS Topical Meeting on Probabilistic Safety Methods and Applications, Volume 2, February 1985, San Francisco, CA.

6. G. Edison, "Sizewell-B: Analysis of British Application of U.S. PWR Technology," NUREG-0999, May 1983.

7. H. L. Schnurer and H. G. Seipel, "The Safety Concept of Nuclear Power Plants in the Federal Republic of Germany," Nuclear Safety, 24, 743 (1983).

8. ATWS Final Rule - Code of Federal Regulations, Title 10, Section 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram Events for Light-Water Cooled Nuclear Power Plants," June 1984.

9. R. Youngblood et al., "Fault Tree Application to the Study of Systems Interactions at Indian Point 3," Brookhaven National Laboratory, NUREG/CR-4207, January 1986.

10. American Nuclear Society and Institute of Electrical and Electronics Engineers, "A PRA Procedures Guides," NUREG/CR-2300, January 1983.

11. R. B. Worrell and D. W. Stack, "A SETS User's Manual for the Fault Tree Analyst," Sandia National Laboratories, NUREG/CR-0465, SAND77-2051, November 1978.

12. "Reactor Risk Reference Document," U.S. Nuclear Regulatory Commission, NUREG-1150, Draft for Comment, February 1987.

13. W. R. Cramond et al., "Shutdown Decay Heat Removal Analysis of a Combustion Engineering 2-Look Pressurizer Water Reactor," Sandia National Laboratories, NUREG/CR-4710, August 1987.

Table 4.1 Criteria for PWR Ice-Condenser Containment
Guideline 1: Maintain Containment Integrity

Concern: SARRP predicts containment failure near the time of vigorous core-concrete interaction. Thus a substantial fraction of the vaporization release may be released from containment.

Functions: Containment Spray (Guideline 1.A)
- Containment Heat Removal
- Fission Product Scrubbing
- Debris Bed Cooling

Filtered Venting (Guideline 1.B)

Guideline 1.A. Provide Containment Spray

Basis: Early initiation of containment spray will help flood the reactor cavity and cool the core debris as it exits from the reactor vessel. Continued spray should help control the containment pressure rise because of the decay heat load and should aid in fission-product decontamination.

Caution: Failure of recirculation has been identified as a dominant contributor to core damage. For small break LOCA sequences, early initiation of core spray will rapidly deplete the refueling water storage tank (RWST) and will limit the available RCS injection capability if recirculation is unsuccessful (See Guideline 5.A.).

Criteria:

The following should be assessed to ensure containment heat removal capability:

1.A.1. The heat removal provided by the containment spray related components should be sufficient to remove heat loads anticipated during the dominant accident sequences. These loads include but are not limited to decay heat and the chemical energy released from metallic oxidation.

1.A.2. Containment spray should commence when the containment pressure exceeds a predetermined value calculated in accordance with the appropriate PWR emergency response guidelines (ERGs) on a consistent basis with the Final Safety Analysis Report (FSAR) requirements.

1.A.3. Containment spray should be terminated when the containment pressure decreases below a predetermined value calculated in accordance with the appropriate ERGs on a consistent basis with the FSAR requirements.

Guidance: Alternate sources of containment spray such as a diesel-driven fire pump should be assessed for their capability to provide sufficient flow and head for adequate containment heat removal.

1.A.4. The ability of equipment designated for containment spray to function in a reliable manner under the predicted containment conditions associated with sequences for which operation of the containment spray is needed should be assessed.

Table 4.1 (Continued)

1.A.5. Operator training and emergency procedures should specify the flow paths and specific components to be aligned and their required positions for initiating containment spray. If a backup system and/or equipment are to be utilized, operator training and procedures should identify the flow paths and specific actions required for any temporary system cross connections.

1.A.6. Operator training and emergency procedures should specify the plant parameters that will prompt the operators to initiate and terminate the containment spray. Training and procedures should be consistent with the time required to align the system and components as required.

The following should be assessed in order to evaluate the capability of the containment spray to reduce fission-product contamination of the containment atmosphere:

1.A.7. The spray system should be assessed for its ability to cover the entire containment volume (upper and lower compartments) for an adequate time, with spray droplets of an appropriate size. Such an assessment would include the total amount of water available for long-term spray operation, as well as the pressure under which the water could be supplied to the spray headers by various sources. The elevation of the spray, the nozzle spray pattern, as well as large obstructions below the spray headers, should be considered when assessing volume coverage.

In addition, to promote debris bed cooling the following criteria should be assessed to ensure flooding of the reactor cavity before RPV failure:

1.A.8. The spray initiation point (Criterion 1.A.2) should be assessed to ensure that the sprays will be initiated early enough to flood the reactor cavity before RPV failure for the dominant accident sequences.

1.A.9. The spray termination point (Criterion 1.A.3) should be assessed to ensure that spray termination will not allow the debris bed to reheat after it has been successfully cooled..

Guideline 1.B. Provide Filtered Venting

Basis: For those plants where ac independent containment sprays and long term containment heat removal are not available venting should be considered to ensure fission-product mitigation for late containment failure events.

Caution: Containment venting should not be indiscriminantly performed. A clear understanding of the accident sequence in progress should have been attained prior to initiating venting. The effects of venting should have been assessed and made known to the operators during the training program. The assessment should include the effects of containment venting on the operation of emergency core cooling (ECC) injection systems and health consequences.

Table 4.1 (Continued)

Criteria:

The following should be assessed to ensure filtered venting capability:

1.B.1. For accident sequences where filtered venting has been assessed to be beneficial, filtered venting should commence when containment pressure reaches the predetermined containment venting pressure set point. In selecting the containment venting pressure set point, the following functions should be assured:

a. the ultimate containment pressure capability would not be exceeded.

b. the backpressure acting on the safety relief valve assemblies would not prevent them from performing their function.

c. the vent valve assemblies would not be prevented from performing their function.

1.B.2. If local initiation of filtered venting is necessary, the time required to perform this function and the potential for exposing personnel to the harsh environment should be taken into account in the training and procedures.

1.B.3. Operator training and emergency procedures should specify the plant parameters that will prompt the operators to make preparation, commence and terminate the venting sequence. The training and procedures should also be consistent with the required actions and timing of those actions so venting will commence immediately when required (see Criteria 1.B.1 and 1.B.2). The training and procedures should further specify the flow path(s) available for venting, specific components to be aligned, and the required positions/states for these components. The training and procedures should specify how to proceed if termination of venting is not possible.

1.B.4. For each accident sequence where venting is credited (e.g., assumed to prevent containment failure) the capacity of the vent lines and associated vent valves should be assessed to determine whether the venting capacity has the capability to decrease containment pressure.

1.B.5. The filtering media should be capable of reducing the released fraction of the radioactivity of the non-noble gas component by an order of magnitude. For some plants containment sprays and/or natural deposition processes (i.e., late in the accident sequence) may provide sufficient decontamination to provide "filtering."

1.B.6. The capability of equipment used to support venting to function under predicted environmental and fluid loads associated with severe accident sequences (including station blackout) should be assessed to determine whether the equipment would be available for an appropriate time of operation including the vaporization release phase of core-concrete interaction.

Table 4.1 (Continued)

1.B.7.	The filters should be capable of accepting the aerosol loading from core-concrete interactions while remaining functional.

Table 4.2 Criteria for PWR Ice-Condenser Containment
Guideline 2: Ice-Condenser System (ICS)
and Air Return Fan System (ARFS)

Concern: Design features of the ICS and ARFS allow for a lower containment design pressure and a smaller containment volume than those for an equivalent size PWR with a large-volume containment. The ICS works as a pressure suppression and radioactivity scrubbing device. The ARFS enhances the effectiveness of the ICS and promotes mixing of hydrogen with air in the containment. Failures of these systems (including bypass of these systems) would lead to rapid pressurization of the the containment because of steam or noncondensable gas buildup and hydrogen combustion. This rapid pressurization could result in early containment failure with little effective fission-product scrubbing.

Functions: Pressure Suppression
Fission-Product Scrubbing
Containment Air Circulation (Guideline 2.A.)

Guideline 2.A. Provide Reliable Operation of ICS and ARFS

Basis: Implementation of the following criterion will significantly improve reliability of the systems thus reducing the risk associated with their failures.

Criterion:

2.A.1. The capability of equipment designated for the ICS and ARFS to function under predicted environmental and fluid loads associated with dominant severe accident sequences (including station blackout) should be assessed to determine whether the equipment would be available for an appropriate time of operation.

Table 4.3 Criteria for PWR Ice-Condenser Containment
Guideline 3: Hydrogen Control

Concern: SARRP indicates that there is a substantial probability of early containment failure because of uncontrolled hydrogen combustion for station blackout (SBO). Rapid releases of hydrogen and core debris during high-pressure meltdowns also contribute to early containment failure. Enhancement of depressurization capability is addressed in Guideline 8.

Functions: Hydrogen Control

<u>Guideline 3.A. Provide Reliable Hydrogen Igniters</u>

Basis: Implementation of the following criteria will improve the operability of the igniters for all accident sequences thus reducing the likelihood of early containment failure.

Criteria:

3.A.1. The capability of the hydrogen igniters to function under predicted environmental and fluid loads associated with the dominant severe accident sequences including SBO sequences should be assessed to determine whether the equipment would be available for an appropriate time of operation.

3.A.2. In order to ensure that localized high concentrations of hydrogen do not occur, the capability of the air return fan system (ARFS) to function under predicted environmental and fluid loads associated with the dominant severe accident sequences (including SBO) should be assessed to determine whether equipment would be available for an appropriate time of operation.

Guidance: For individual plants which are found to have a high SBO core-damage frequency (CDF), ac independent igniters and ARFS should be considered.

Table 4.4 Criteria for PWR Ice-Condenser Containment
Guideline 4: Interfacing Systems LOCA
(Including Steam Generator Tube Rupture)

Concern: The interfacing systems LOCA sequences represent potentially high-release sequences and were found to contribute significantly to risk, even though their contribution to core-damage frequency (CDF) is not significant. The contribution to CDF by the steam generator tube rupture (SGTR) sequences is also relatively small. However, the associated risk can be potentially significant because the leakage of primary reactor coolant to the secondary side of the steam generator (SG) can cause release of radioactivity to the outside of containment via the steamlines, the turbines, the condenser and the SG atmospheric steam dump and safety valves.

Function: Maintain Primary System Integrity (Guidelines 4.A and 4.B)

Guideline 4.A. Prevent Overpressurization of Low-Pressure Systems

Basis: Implementation of the following criteria will ensure the frequency of an interfacing systems LOCA will remain acceptably low.

Note: Resolution of the generic issue (GI-105) which deals with interfacing systems LOCAs for both BWRs and PWRs, may impact this guideline. Therefore, the criteria below should be considered as appropriate pending resolution of the generic issue.

Criteria:

4.A.1. All low-pressure lines that potentially could be overpressurized should be identified and should be provided with alarms to alert the operator of an overpressure event.

4.A.2. The equipment designated to provide isolation and prevent overpressurization, such as the RHR line isolation valves or the low pressure injection system check valves, should periodically undergo operability testing and local leak rate testing (LLRT).

4.A.3. The relief capability of the relief valves designated to mitigate low-pressure system overpressurization should be established. In most, if not all, cases these relief valves were not sized with the possibility of an interfacing systems LOCA in mind. However, given that an interfacing systems LOCA occurs in a non-isolatable portion of a low pressure system, there may be alternatives available to the operator such as taking advantage of additional relief valves. If such or similar actions are found to be helpful, they should be factored into the appropriate emergency procedures.

4.A.4. Operator training and procedures should specify the actions to be taken to isolate the low-pressure systems identified above or depressurize the primary system, thereby mitigating the consequences of the interfacing systems LOCA.

4.A.5. After each reactor shutdown and cooldown, testing of the isolation function of the pressure isolation valves should be performed. Testing of these valves should not be performed under reactor operating conditions.

Guideline 4.B. Prevent Steam Generator Tube Rupture (SGTR) and Minimize Its Consequences

Basis: The following criteria are provided to ensure that the risk because of SGTR events at each plant is a small fraction of the overall risk (See NUREG-0844).

4.B.1. The structural integrity of steam generator (SG) tubes should be maintained by protecting them against all plausible causes leading to weakening, cracking or bursting of the tubes. As part of the resolution of the unresolved safety issues related to SGTR (USIs A-3, A-4 and A-5) the NRC staff has issued Generic Letter 85-02 to ensure that each licensee's steam generator tube integrity program complies with staff recommendations.

4.B.2. Operator training and procedures should specify the means by which the occurrence of SGTR can be correctly diagnosed, the faulty SG identified and isolated in a timely manner, and the necessary recovery actions to be taken. These actions include reactor coolant system (RCS) cooldown to below the saturation temperature corresponding to the faulty SG pressure by dumping steam only from the intact SGs to establish sufficient subcooling margin, subsequent RCS depressurization to the faulty SG pressure with establishment of sufficient RCS inventory and safety injection (SI) termination to avoid repressurization and further primary-to-secondary leakage.

4.B.3. If offsite power is unavailable during the recovery actions, the RCS cooling described under 4.B.2 above should be achieved by using the atmospheric relief valves on the intact SGs, since neither the turbine bypass valves nor the main condenser would be available. Also, RCS pressure should be controlled by using pressurizer PORVs or auxiliary spray since normal pressurizer spray will be unavailable because of loss of the reactor coolant pumps (RCPs).

Guidance: The effectiveness of various accident management strategies in reducing overall risk for specific plant types is being investigated by an NRC/RES program. These operator related criteria (e.g., 4.B.2 and 4.B.3) are provided to emphasize the importance of the operators' actions in mitigating severe accidents. These criteria are considered to be appropriate pending NRC resolution of accident management issues.

Table 4.5 Criteria for PWR Ice-Condenser Containment Guideline 5: Operator Response for Recirculation

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Concern: Inadequate decay heat removal because of failure in the recirculation switchover operation has been found to be a leading contributor to severe accident sequences, in particular, sequences initiated by small LOCAs. Failure in the recirculation switchover of the high pressure systems leads to core damage because of insufficient coolant makeup and thus inadequate decay heat removal from the core. Failure in the recirculation switchover of the containment spray system and failure to establish a containment spray path from RHR lead to inadequate containment heat removal (CHR). Containment failure because of inadequate CHR could create net positive suction head (NPSH) problems for the pumps taking suction from the containment sump, which could in turn lead to recirculation failure and subsequent core damage.

Functions: Adequate Recirculation Cooling (Guideline 5.A)
Adequate Containment Heat Removal (Guideline 5.B)

Guideline 5.A. Provide Adequate Recirculation Cooling

Basis: Implementation of the following criteria will significantly reduce the CDF and risk because of core damage associated with the recirculation cooling phase.

Criteria:

5.A.1. Operator training and procedures should specify the actions to be performed to realign flow paths, including recovery actions that must be performed during recirculation switchover (especially at high RCS pressure) and RHR heat exchanger injection cooling. These actions should be specified in an integrated fashion, so that no single operator has a designated role whose incorrect action would cause loss of the switchover or cooling functions.

5.A.2. Training and procedures should specify that the upper compartment drain plugs should be removed after refueling to ensure that core inventory recirculation and containment spray recirculation are not adversely affected. They should also specify that the containment sump strainer should be free of plugging.

5.A.3. Plant response to small LOCAs should be investigated to see if there are measures that can be taken to delay or obviate the need to enter the recirculation phase. For example, to delay the emptying of the RWST by delaying initiation of containment spray via raising its setpoint. Note that any change in spray setpoint must be done on a consistent basis with requirements in the FSAR.

5.A.4. Training and procedures should attempt to avoid failure of both injection pumps because of switchover to a sump which does not provide sufficient NPSH to operate the pumps. Consideration should be given to providing procedures which will specify refill of the RWST to provide continued primary injection capability if recirculation switchover fails.

Guideline 5.B. Provide Adequate Containment Heat Removal (CHR)

Basis: Implementation of the following criteria will significantly reduce the risk associated with containment failure.

Criteria:

5.B.1. Operator training and procedures should specify all actions (including alternatives) required to initiate and maintain CHR under severe accident conditions when recirculating from the containment sump.

5.B.2. When manual-remote operation of equipment is required to initiate CHR, the operator training and procedures should specify the actions for performing these functions and provide an understanding of and experience with the time required to perform them.

5.B.3. The capability of equipment designated for CHR to function under predicted environmental and fluid loads associated with dominant severe accident sequences (including SBO) should be assessed to determine whether the equipment would be available for an appropriate time of operation. Note that this criterion does not imply that the equipment will be qualified to a new design envelop.

Table 4.6 Criteria for PWR Ice-Condenser Containment
Guideline 6: Station Blackout

Concern: Station blackout (SBO) sequences have been shown to be one of the leading classes of severe accident sequences in terms of both CDF and risk.

Function: Reliable Operator Response (Guideline 6.A)
Post Accident Decay Heat Removal Capability (Guideline 9.A)

Guideline 6.A. Provide Operator Response During Station Blackout

Basis: Significant study and research have preceded current work on severe accidents; in particular, reference is made to the rulemaking activity already under way on SBO. It is assumed that when the SBO rule is finalized, some requirements of the rule may be similar in form to the criteria below. Nevertheless, during an individual plant examination (IPE), it is important to highlight those areas which previous PRAs found to be important contributors to the SBO CDF. In addition, auxiliary feedwater system (AFWS) turbine-driven pump reliability is being addressed in GI-124 and the following related criteria should be considered appropriate pending resolution of this generic issue.

Criteria:

6.A.1. Operator training and procedures should specify the actions that should be performed to provide core cooling and decay heat removal and should specify the systems and components to be aligned and their required positions.

6.A.2. Operator training and procedures should specify the actions required to restore offsite and onsite emergency ac power prior to depletion of the dc power supply during SBO.

6.A.3. Operator training and procedures should specify the actions that should be performed to shed nonessential loads requiring dc power to increase the length of time that the plant can cope with an SBO.

6.A.4. Operator training and procedures should specify any actions that should be taken prior to the loss of dc power given that ac power has not been restored.

6.A.5. Operator training and procedures should specify actions required to initiate operation and/or assure that containment spray and hydrogen igniters are operable under SBO conditions. If containment spray or igniters are lost under degraded core conditions, specific procedures should address the conditions under which spray or igniters can be restored.

Table 4.6 (Continued)

6.A.6.	The capability of a turbine-driven auxiliary feedwater train or its equivalent to function under predicted environmental and fluid loads associated with severe accident sequences (including SBO sequence) should be assessed to determine whether the equipment would be available for an appropriate time of operation.
6.A.7.	Special emphasis should be placed upon the review of the AFWS to assure that common cause failures have been eliminated to the extent practical from the design.

Table 4.7 Criteria for PWR Ice-Condenser Containment
Guideline 7: Component Cooling Water

Concern: An event initiated by loss of component cooling water (CCW) or essential service water (ESW) may lead to reactor coolant pump (RCP) seal LOCA sequences because of loss of RCP seal cooling and seal injection, under the condition that high-pressure injection also fails. This is an important contributor to CDF at Sequoyah, as is also the case at other PWRs. RCP seal and CCW failure are currently designated as generic issues by the NRC (GI-23 and GI-65, respectively).

Function: Adequate Cooling of Engineered Safety Features Systems (Guideline 7.A)

Guideline 7.A. Provide Adequate Cooling to Engineered Safety Features Systems

Basis: Implementation of the following criteria on CCW will significantly reduce the CDF and risk because of RCP seal LOCA sequences. Loss of CCW initiated by loss of electric power is addressed by Guideline 6: Station Blackout. If, depending on the design configuration of a particular plant, loss of ESW is likely to result in an RCP seal LOCA, then the criteria should be applied to the ESW system.

Criteria:

7.A.1. Operator training and procedures should specify actions to be taken to shed nonessential loads requiring CCW (or ESW) to increase the length of time that the plant can cope with partial loss of CCW (or ESW).

7.A.2. Special emphasis should be placed upon the review of the CCW (or ESW) system to assure that common cause failures have been eliminated to the extent practical from the design.

7.A.3. The capability of the CCW (or ESW) system and associated equipment to function under predicted environmental and fluid loads associated with severe accident sequences should be assessed to determine whether the equipment would be available for an appropriate time of operation.

7.A.4. Alternate injection systems to cool the RCP seals or improved seal designs which prevent leakage should be considered for those plants where RCP seal LOCAs comprise a dominant part of the CDF.

Table 4.8 Criteria for PWR Ice-Condenser Containment
Guideline 8: Reactor Coolant System (RCS)
Depressurization by Secondary Blowdown

Concern: Small or medium LOCAs followed by failure of high-pressure injection system (HPIS) or high-pressure recirculation system (HPRS) were determined to be important contributors to CDF for the plant under review. Some of these transients were also found to contribute quite significantly to risk.

Function: Depressurize the RCS (Guideline 8.A)

Guideline 8.A. Provide RCS Depressurization Capability

Basis: Implementation of the following criteria will enhance the core inventory makeup capabilities of the low-pressure emergency coolant injection and recirculation function in the event that the high-pressure systems are unavailable.

Criteria:

8.A.1. Operator training and procedures should specify the actions to be performed to realign flow paths, initiate and control RCS depressurization by secondary blowdown (blowing down the secondary side of the SG) in order to provide emergency core makeup via the low-pressure injection systems (LPIS) if the high-pressure systems are unavailable. The procedures, such as opening the SG atmospheric steam dump valves and maintaining auxiliary feedwater or main feedwater to the SG, should follow the instructions given in the appropriate emergency response guidelines and account for the possibility of steam generator tube rupture (SGTR--see Guideline 4.B).

8.A.2. The capability of equipment and systems designated to depressurize the RCS by secondary blowdown to function under predicted environmental and fluid loads associated with severe accident sequences should be assessed (this does not imply that the equipment will be qualified to severe accident conditions) to determine whether the equipment would be available for an appropriate time of operation.

Guidance: The effectiveness of various accident management strategies in reducing overall risk for specific plant types is being investigated by an NRC/RES program. These operator related criteria (e.g., 8.A.1) are provided to emphasize the importance of the operators' actions in mitigating severe accidents. These criteria are considered to be appropriate pending NRC resolution of accident management issues.

Table 4.9 Criteria for PWR Ice-Condenser Containment
Guideline 9: Reactor Coolant System (RCS)
Feed & Bleed Cooling

Concern: RCS feed & bleed cooling can be employed to remove the decay heat in case RCS heat removal via the SGs fails completely. If the RCS feed & bleed cooling also fails, core uncovery would ensue.

Functions: Provide Alternate Core Heat Removal Capability (Guideline 9.A)

Guideline 9.A. Provide Operator Response for RCS Feed & Bleed Cooling

Basis: Fulfilling the following criteria will improve the chances for successful RCS heat removal subsequent to loss of main and auxiliary feedwater systems. Overall decay heat removal capability is being addressed by USI A-45. Note that feed and bleed capability has been estimated[13] to reduce CDF for one PWR by a factor of three.

Criteria:

9.A.1. Assess the capacity of the power operated relief valves (PORVs) or head vent valves and the HPIS to remove decay heat.

9.A.2. Operator training and emergency operating procedures should specify the actions required for RCS feed & bleed cooling. In particular, the diagnostic means of determining the need for RCS feed & bleed cooling and the detailed procedures to be followed as well as the number of PORVs to be manually opened should be clearly stipulated. Appropriate means of containment heat removal should also be available (Guideline 5.B).

Table 4.10 Criteria for PWR Ice-Condenser Containment
Guideline 10: Anticipated Transients
Without Scram (ATWS)

Concern: Although PRAs for some PWRs indicate that ATWS sequences are not significant contributors to CDF, they have been studied extensively in the past because they may lead to plant conditions that may render several engineered safety features ineffective and thus may contribute significantly to risk.

Function: Reliability of the Reactor Protection System and Operator Response During ATWS (Guideline 10.A.)

Guideline 10.A. Provide Operator Response During ATWS

Basis: The criteria developed here are based on the assumption that each of the plants is (or will be) in compliance with the ATWS final rule dated July 26, 1984. PRA studies have shown that the predicted CDF because of ATWS is significantly lowered based upon modifications which comply with the ATWS rule. The major thrust of the ATWS rule is on the addition and/or upgrading of scram related systems and equipment to prevent an ATWS. NUREG-1150 point to potential benefits for improved operator training and procedures to mitigate the effects of an ATWS and prevent core damage from occurring. For any individual plant which may be found to be vulnerable to ATWS, the following criteria reflect added measures that emphasize the operator's role and function in mitigating an ATWS initiator.

In the case when the automatic scram system fails during plant transients, the operator is required to attempt to manually scram the reactor and inject borated water to the RPV by using the charging system including repositioning the valves which are necessary to properly align the charging system to the boron supply.

Criteria:

10.A.1. Operator training and procedures should specify the actions required for the manual scram and emergency boration. They should also specify the responsibilities of operating staff crew members and clarify how information will be exchanged among them. In particular, instrumentation readings may have to be relayed between the crew members(s) operating the control boards and the senior reactor operator coordinating the crew's response to accident.

10.A.2. The capability of systems and equipment designated for operation during an ATWS to function under predicted environmental and fluid loads associated with severe accident sequences should be assessed to determine whether the equipment would be available for an appropriate time of operation.

Table 4.11 Criteria for PWR Ice-Condenser Containment
Guideline 11: Support System Interdependencies

Concern: When conducting a probabilistic risk assessment (PRA), individual plant evaluation (IPE) or similar analysis, it is imperative that the support system interdependencies be fully developed, understood and reflected in the final results. Otherwise there is no assurance that the dominant core-damage/risk sequences have been identified.

Function: Support System Interdependencies (Guideline 11.A)

Guideline 11.A. Examine Support System Interdependencies

Basis: Implementation of the following criteria will ensure that the full set of support system interdependencies have been identified and have been reflected in the results.

Note: The following criteria are easily outlined but are not easily implemented. The complex nature of a nuclear power plant makes it imperative that this area of analysis be fully examined. However, since no two plants have identical support systems, this analysis should be done on a plant specific basis.

Criteria:

11.A.1. All systems that provide any direct support to either a frontline or support system should be identified along with its supported system. For each dependency that is identified, the failure mechanism and time should be estimated.

11.A.2. Each dependency should be conditioned as appropriate as to what sequences or under what (if not all) circumstances it applies. In view of its importance, a separate station blackout dependency table should be provided which gives the available systems, their anticipated survival period and the ultimate cause (e.g., no room cooling) of their failure.

11.A.3. As in Criterion 11.A.2, a dependency table should be provided which shows the effects of loss of component cooling water (or service water) on other systems.

11.A.4. The dependencies should then be linked together (preferably by computer) within the analysis in order that the extent to which their influence reaches through the systems to a consequence will be discovered. This will help identify secondary dependencies to ensure that no one failure in a support system has any unknown critical outcome on other support or front line systems.

Table 4.12 Criteria for PWR Ice-Condenser Containment Guideline 12: Flooding of Emergency Equipment

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Concern: An excessive water release into a portion of the turbine building or auxiliary building outside the primary containment which houses a concentration of safety-related equipment raises the possibility of a common-mode event disabling key equipment. At least one PWR plant has been identified in which the location of safety equipment makes the internal flooding initiator a substantial contributor to CDF.

Function: Prevent or Mitigate Internal Flooding (Guideline 12.A)

Guideline 12.A. Prevent or Mitigate Internal Flooding

Basis: Implementation of the following criteria may reduce the potential of a common-mode failure of safety equipment because of internal flooding for PWR plants where the systems' layout combines important safety equipment in compartments with exposure to possible inundation.

Criteria:

12.A.1. For those areas identified as vulnerable to flooding, instrumentation should be added to alert the operator of such an occurrence.

12.A.2. Operator training and procedures should ensure that the operator will respond to isolate any internal floods that occur.

12.A.3. Operator training and procedures should ensure that the operator is prepared to use alternate injection sources still available if flooding causes a common-mode failure of ECCS equipment.

12.A.4. The electrical systems should be assessed for the possibility of cascading failures because of flood-induced electrical shorts. Additional isolation devices (e.g., circuit breakers) should be considered, if needed.

12.A.5. The addition of water-tight doors/walls might be effective ways to lessen the common-mode threat. Water-tight doors between compartments should be alarmed to notify the operator when they have been left open.

APPENDIX A

SEVERE ACCIDENT RISK INSIGHTS

This appendix reviews the Industry Degraded Core Rulemaking Program (IDCOR) and the Nuclear Regulatory Commission (NRC) Accident Sequence Evaluation Program (ASEP) analyses for the Sequoyah plant which is a PWR with an condenser containment. Differences between the studies are identified and insights are provided which helped in the development of the plant type-specific guidelines and criteria for preventing and mitigating severe accidents which are discussed in Section 4 of this report. In addition to the review of the IDCOR and ASEP analyses for Sequoyah, design features of other plants (D.C. Cook, McGuire, Catawba, and Watts Bar) with an ice-condenser containment are also overviewed to identify similarities and differences among the plants. The insights gained from this overview of the design features also contributed to the development of the guidelines and criteria in Section 4.

The Sequoyah Nuclear Plant Units 1 and 2, located near Chattanooga, in Hamilton County, Tennessee, are Westinghouse PWRs of thermal output of 3411 MW per unit.

Each unit of the Sequoyah Nuclear Plant has an ice-condenser system (ICS) for containment cooling and energy absorption in the event of an accident. The air return fan system (ARFS) enhances the ICS by circulating air between the lower and upper containment compartments through the ice-condenser compartment, and dilutes hydrogen concentrations in potentially stagnant regions of the containment by ensuring continuous flow of air. These two design features, i.e., ICS and ARFS, allow for a lower containment design pressure and a smaller containment volume than those for a reactor with a large-volume containment.

Another design feature at Sequoyah is the upper-head injection system (UHIS), which operates automatically during the injection phase of loss-of-coolant accident (LOCA) scenarios to provide an initial high flow of borated cooling water into the top of the reactor vessel. The UHIS supplements the cold leg accumulator system (CLAS) but is not considered necessary for successful mitigation of LOCAs.

Table A.1 summarizes the design of the safety-related systems at Sequoyah in comparison with those at D.C. Cook, McGuire, Catawba, and Watts Bar. Figure A.1 shows a schematic for the Sequoyah engineered safety features and Figure A.2 shows a heat and fluid flow diagram for Sequoyah's various cooling systems. Figure A.3 and Figure A.4 show a schematic of the Sequoyah ICS and a simplified flow diagram of the Sequoyah ARFS, respectively.

Section A.1 describes the review of the various estimates of the core damage profile. Core-meltdown phenomena and containment response are addressed in Section A.2. Differences between IDCOR and the NRC Severe Accident Risk Reduction Program (SARRP) estimates of fission product release and offsite consequences are discussed in Sections A.3 and A.4. Finally, Section A.5 summarizes the insights gained from the review of these studies.

A.1 Core-Damage Profile

The main objective of this section is to present the Sequoyah core-damage profiles emerging from the IDCOR[1] and ASEP/SARRP[2,3] analyses.

The primary information reviewed in this section is from Refs. 1 through 3. The information contained in Refs. 4 and 5 was also used in the review.

A.1.1 IDCOR Assessment of Sequoyah Core-Damage Frequency (CDF)

Three different sets of Sequoyah risk profiles are presented in Section 7 of the IDCOR report[1] including the pre-IDCOR risk profile, the IDCOR baseline risk profile, and the committed risk profile. The pre-IDCOR risk profile is based primarily on the results of the Sequoyah RSSMAP Study (NUREG/CR-1659) which was conducted to identify the dominant accident sequences for a PWR ice-condenser plant for comparison with those identified in the Reactor Safety Study (RSS) (NUREG-75/014). The IDCOR baseline risk profile was developed by modifying the pre-IDCOR risk profile to reflect changes in current plant configuration and operation procedures as well as initiating event frequency, success criteria, system unavailability, containment response, and fission-product transport information. It serves as an estimate of current plant risk and CDF which can be used to compare with various safety goals, and also was utilized as a basis for estimating the impact of several potential modifications evaluated in the IDCOR report. The committed risk profile differs from the IDCOR baseline risk profile only in that it takes into account one of the new emergency procedures currently being implemented at Sequoyah, namely, depressurization of the reactor coolant system (RCS) using steam generators (SGs) followed by core inventory makeup employing the low-pressure coolant injection or recirculation system. Risk reduction potential of other planned procedures and plant modifications was not considered.

The initiating event frequencies used by IDCOR to quantify the CDF are summarized in Table A.2. Comparisons of the CDF constituting part of these risk profiles are shown in Table A.3. A brief review and discussion of these CDFs is given in the following sections.

A.1.1.1 Pre-IDCOR Core Damage Frequency

The dominant accident sequences for the pre-IDCOR CDF are failures of core inventory makeup following LOCA events and the interfacing systems LOCA (ISL). Approximately 60% of the CDF is contributed by the recirculation failure sequences, S_2H, S_1H, and S_2HF, and roughly 17% by failures of injection after LOCAs, S_2D and S_1D. The ISL contributes about 8%.

The initiating event frequencies and the point-estimate failure probabilities of various events used in quantifying the pre-IDCOR CDF are essentially based on those listed in Table 7-4 and Table 9-2 of the RSSMAP report (NUREG/CR-1659). Appendix B of that report contains a detailed description of the methodology employed in computing the point-estimate failure probabilities of some of the key events such as H (emergency coolant recirculation system) and D (emergency coolant injection system). Simplified fault trees are also presented in that report.

A.1.1.2 IDCOR-Baseline Core-Damage Frequency

The single, most dominant contributor (79%) to IDCOR-baseline CDF is the sequence, S_2H, i.e., a small LOCA with failure of core makeup during recirculation. As compared with pre-IDCOR, the overall CDF increased by nearly a factor of four, mainly because of the significant increase in the initiating frequency for small LOCA, S_2. In the following discussion, a brief explanation is given with regard to how the IDCOR-baseline CDF was calculated by modifying the pre-IDCOR value for each of the accident sequences.

For convenience of discussion, the function or system unavailabilities used in quantifying the CDF are summarized in Table A.4.

- Sequence S_2H

Although the initiating frequency of a small LOCA event, S_2, is increased by a factor of about 11, the failure probability of Event H, i.e., failure of recirculation, is reduced by roughly a factor of three after incorporating the following changes:

(1) The independent, simultaneous failures of the two sump valves and single failures in the sump system are treated as common cause failures of the low pressure recirculation system (LPRS) and the containment spray system (Event HF).

(2) The switchover to hot-leg recirculation at 24 hours for S_2 LOCAs is assumed not to be required for both LPRS and high-pressure recirculation systems (HPRS).

(3) A nonrecovery factor of 0.23 is used for recovery from failure of one of the sump valves because of electrical actuation faults.

(4) The unavailability of both low-pressure pumps (2.0E-5)* is added.

(5) A nonrecovery factor of 0.1 is used for recovery from room-cooling failures.

(6) A nonrecovery factor of 0.28 is used for recovery from failures of suction valves connecting the LPRS to the high pressure pumps.

The overall impact is an increase in the CDF associated with this sequence by a factor of about four. It must be remarked that the dominant contributor to failure of recirculation, H, continues to be the operator error in switching over from injection to recirculation.

- Sequence S_1H

The initiating frequency for intermediate LOCA, event S_1, is basically the same as in the pre-IDCOR analysis. All the changes listed above under sequence S_2H also apply to this case for reevaluation of H, except that the hot-leg recirculation switchover failure is included with a nonrecovery factor of 0.01. This results in a substantial reduction of both H and the CDF of this sequence.

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*2.0E-5 = 2.0×10^{-5}.

• Sequence S_2D

Despite the increase in the initiating frequency of S_2, the CDF associated with this sequence is reduced because of a large reduction in the failure probability of the coolant injection, Event D, which resulted from the change in the success criteria of D for S_2 LOCAs. The requirements were changed from "one-of-two charging trains and one-of-two safety injection trains" to "any one of the charging or safety injection trains." Failure of the refueling water storage tank (RWST) was added to the sequence, but its contribution is quite small.

• Sequence V

The initiating frequency for ISL (Event V) is reduced by about a factor of five (NUREG-0773) to reflect the increased leak testing and surveillance of the check valves on the low-pressure injection lines.

• Sequence S_2HF

The roughly tenfold increase in the initiating frequency of S_2 is compensated by a factor of ten decrease in the failure probability of the Event HF. The common-cause human error (to remove the upper to lower compartment drain plugs) is reduced by applying a 0.01 human error factor to reflect current plant procedures involving verification. Also, credit for recovery of a sump valve (0.23 nonrecovery factor) was included for S_1 and S_2 sequences. These are primarily responsible for the large reduction in HF frequency.

• Sequence S_1D

For intermediate LOCA, the failure of coolant injection, D, and the initiating frequency remain essentially unchanged from the pre-IDCOR values. Therefore, the same CDF for this sequence is obtained in both studies.

• Sequence $T_{23}ML(Z)$

The initiating frequency of T_{23} is increased from 7 to 8.1. Also, the unavailability of M (main feedwater) is increased by a factor of ten after reviewing current PRAs. Feed and bleed cooling (Event Z), following loss of the auxiliary feedwater system (ATWS), is newly added to the baseline evaluation. The overall impact reduces the CDF of this sequence by a factor of three.

• Sequence S_1HF

The reasons for the decrease in HF have been explained under (e) (Sequence S_2HF). The CDF of this sequence is, thus, reduced by about a factor of ten.

• Sequence AH

There is a significant increase in the initiating frequency of A (large LOCAs). The failure probability of H, however, is reduced by about a factor of ten by incorporating the changes explained for Sequence S_1H, with the exception that no recovery of a sump valve is considered. This results in more than a factor of two increase in the CDF of this sequence.

- Sequence T_1B_3MLB'

For loss of offsite power transient (T_1), an event tree is presented in the IDCOR report (Figure A-20 on page A-37) depicting how the CDF of this sequence is calculated for the baseline case. Note that the IDCOR analysis did not consider the possibility of reactor coolant pump (RCP) seal LOCA.

- Sequence AD

The large increase in CDF of this sequence is essentially because of the increase in the initiating frequency of a large LOCA, A.

- Sequence AHF

The failure probability of HF is reduced by about a factor of three after incorporating the changes explained for Sequence S_2HF, with the exception that no recovery of a sump valve is considered. The CDF from this sequence, nevertheless, increases significantly because of the large increase in the initiating frequency of a large LOCA, A.

A.1.1.3 <u>IDCOR-Committed Core-Damage Frequency</u>

Implementation of the new emergency procedure described earlier allows low-pressure makeup systems to be used for core inventory makeup if high-pressure systems are unavailable, either in injection or recirculation. Only the sequences, S_2H, S_1H, S_2D, and S_1D, are affected by this new procedure, resulting in roughly a factor of five decrease in the total CDF. All the initiating frequencies used in quantifying the IDCOR-committed CDF are identical to those used in the baseline calculations. The basis for the change in H or D from the baseline value is briefly summarized below for each of the four affected sequences.

- S_2H Sequence

The new procedure has significant impact on the dominant contributor (operator failure in the high-pressure recirculation switchover). An operator nonrecovery probability of 0.06 was used for high pressure trains. This change contributes to the reduction of H.

- S_1H Sequence

The causes for reduction in H are identical to those mentioned above for the S_2H sequence.

- S_2D Sequence

A nonrecovery factor of 0.03 was used for operator error to take the required actions. The unavailability of the low-pressure systems, atmospheric dump valves, and AFWS is included (about 2.0E-3 total). This results in a factor of about three reduction in D.

- S_1D Sequence

The reasons mentioned above for S_2D also apply to this sequence.

A.1.2 ASEP Assessment of Sequoyah Core-Damage Frequency

A.1.2.1 Initiating Events

Table A.5 shows the initiating events considered and their annual occurrence frequencies used in ASEP.

A.1.2.2 Event Trees

References 2 and 3 provide the event trees for Sequoyah. An event tree was developed for each initiator shown in Table A.5. The event tree for the loss of a dc bus initiator (T_{DCX}) uses the event tree for the loss-of-offsite power initiator (T_1). The event trees for transients with the power conversion system (PCS) initially unavailable (T_2), and for transients with the PCS initially available (T_3) are of the same structure.

Accident sequences which involve a stuck-open power-operated relief valve (PORV) from transient event trees are transferred to the event tree for the small LOCA (S_2). Anticipated transients without scram (ATWS) sequences are transferred to an ATWS event tree and further developed therein.

In addition to eight accident initiators considered in Ref. 2, Ref. 3 includes loss of component cooling water (CCW) as an accident initiator. The event tree for this initiator is, however, not provided in Ref. 3.

Accident sequences in which containment heat removal function (served by the containment spray recirculation system (CSRS) and the LPRS) is lost but core heat removal function (served by the emergency core cooling (ECC) injection and recirculation systems) is available are tentatively flagged as "core vulnerable" in Ref. 2. These core vulnerable sequences would be further analyzed to determine whether they eventually lead to core damage. CDF resulting from these core vulnerable sequences are given in Ref. 3 as 13% of the vulnerable sequences.

Each accident sequence is assigned to a plant-damage state depending on the initiator, containment status regarding whether it is intact, failed before core damage, or failed after core damage, and spray status for radioactivity removal. There are 14 plant-damage states in total, including a plant-damage state to which the interfacing systems LOCA (V) sequence is assigned.

A.1.2.3 Fault Trees

Fault trees are developed for systems which constitute the functions in the event tree headings. The fault trees or Boolean equations necessary for event tree quantification are provided in Ref. 3.

A.1.2.4 Data

In general, the failure probabilities used by ASEP[3] are generic data. Data used for diesel-generator failure to start and for valve plugging are, however, said to be plant-specific values for Sequoyah. Beta factors for diesel generators, motor-operated valves, and motor-driven pumps are generic values from Ref. 6 assumed to represent the 95% upper confidence bound.

Operator actions identified in the fault trees and recovery actions in the accident sequence quantification were evaluated using the generic ASEP methodology and human error probabilities.

A.1.2.5 Results

The sequences whose point estimates are greater than 10^{-9}/reactor year are retained in the ASEP study. The results are mean-point estimates, which imply that all the basic event probabilities put in the fault trees and event trees are mean values. In the event tree quantification, the rare-event approximation and the DELETE TERM procedure in the SETS code[7] are used. The DELETE TERM procedure is a way to incorporate systems success in an accident sequence. The results of uncertainty analysis and sensitivity studies are provided in Ref. 3. Uncertainty associated with statistical uncertainty in the failure data was treated by Monte Carlo sampling codes and uncertainties associated with assumptions and modeling were handled by performing several sensitivity studies. The results were presented in "box and whisker" type construction.

Ref. 2 provides the frequencies of the dominant core-damage and core-vulnerable sequences before the common cause failures (using β-factors) and the operator recoveries are incorporated. However, it also provides additional results for three S_2 sequences which incorporated the β-factors and recoveries. Ref. 3 incorporated the common cause failures and the operator recoveries in the dominant CDFs. Note that the approach of the ASEP study is to incorporate the common cause failures (using beta factors) and the operator recoveries at a cutset level after the cutsets are generated. Ref. 3 also provides quantification of the core-damage sequences which result from the core-vulnerable sequences identified in the event trees in Ref. 2. These results are summarized in Table A.6, followed by the nomenclature for the sequences.

The most dominant core-damage sequence in the ASEP study (Ref. 3) is S_2H_2 characterized by the small LOCA initiator and failure of the high-pressure recirculation function. This sequence contributes 34% (2.9E-5) to the total CDF of mean-point estimate 8.6E-5. Main contributors are operator errors involved in the recirculation switchover which requires realignment of several valves, and common cause failure of high-pressure pump suction valves.

Loss of the component cooling water (CCW) system, as an initiator (T_{CCW}), leading to RCP seal LOCA with ECC system failure, is next in importance. This sequence contributes 31% (2.7E-5) to the total CDF, with common cause failure of pumps as a dominant contributor.

S_2H_3, which is characterized by the small LOCA initiator and failures of low-pressure injection in miniflow mode and low-pressure recirculation in support of high-pressure recirculation, contributes 11% (9.7E-6) to the total CDF. S_2H_3F, in which failure of the containment spray recirculation function accompanies S_2H_3, contributes 9% (7.7E-6).

The loss-of-offsite power initiator (T_1) contributes 5% (4.1E-6) to the total CDF. Its main contributing sequences are $T_1D_3WD_1F$ (2.9E-6) and $T_1L_1D_1F$ (8.3E-7). $T_1D_3WD_1F$ is a pump seal LOCA sequence accompanied by failure of the

containment-spray recirculation function in station blackout (SBO). $T_1L_1D_1F$ is a battery depletion sequence accompanied by failure of the containment-spray recirculation function in SBO.

Note that the results of Ref. 3 were used for identification of the most important sequences. This choice was made because Ref. 3 presents the most recent results for Sequoyah from ASEP.

A.1.3 Comparison Between IDCOR and ASEP

Table A.7 compares initiator frequencies used in the IDCOR[11] and ASEP[3] studies. Table A.8 shows the results comparing dominant sequences at the core-damage level for Sequoyah in comparison to results from several other PRAs for PWRs. Note that the results from IDCOR are "mixed"-point estimates, resulting from using mean initiator frequencies and median basic event probabilities, while the results from ASEP are mean-point estimates.

A.1.4 Discussions

The CDF for Sequoyah, assessed in IDCOR,[11] is based upon the results of the Reactor Safety Study Methodology Applications Program (RSSMAP) for Sequoyah Unit 1, which was a simplified "survey and analysis." The RSSMAP study has been updated to reflect changes in current plant features and operational procedures, as well as more recent failure data and initiating event frequencies. It was, thus, a limited study that tried to infer a plant specific risk profile at Sequoyah from a simplified analysis which in turn was done for a different plant (Surry). The grouping of transient initiators in only two categories (loss-of-offsite power and all transients other than loss-of-offsite power) in IDCOR does not allow accurate quantification for sequences which involve modeling of the system failures conditioned on specific accident initiators. This grouping may lead to failure to uncover important sequences (e.g., sequences initiated by loss of component cooling water or loss of a dc bus) which were identified as important contributors to CDF and plant risk in several recent PRAs, including the ASEP study for Sequoyah. Such support system failures may be even more important in other ice-condenser plants for which the support systems configuration is different (see Table A.1).

The ASEP study for Sequoyah appears to be a plant-specific detailed analysis employing the state-of-the-art in PRA methodology. It develops plant-specific event trees for a reasonably complete set of accident initiators and fault trees which are necessary for quantification of the event trees. It is noteworthy that it uses the fault tree linking method for event tree and fault tree evaluation. The data used is, however, less than plant-specific. Most of the failure probabilities and β-factors for common cause failures are generic data except for the probabilities of diesel generator failure to start and valve plugging.

As is shown in Table A.8, the small LOCA initiator (S_2) is the largest contributor to core damage at Sequoyah both in IDCOR analysis and in the ASEP study. Dominant sequences associated with this initiator are S_2H and S_2HF in IDCOR notations, and S_2H_2, S_2H_3, S_2H_2F, and S_2H_3F in ASEP notations. S_2H corresponds to S_2H_2 + S_2H_3 and S_2HF to S_2H_2F + S_2H_3F. Major contributors to these sequences are related to the recirculation switchover involving operator

failures in several steps, and the common cause failures of sump suction valves and high-pressure pump suction valves. The recirculation switchover requires several operator actions to realign flow paths: changing positions of several valves downstream of the low pressure injection pumps, isolation of ECC system from the RWST suction, switchover of containment spray from injection to recirculation, and valving in CCW to the residual heat removal (RHR) heat exchangers. The ASEP study treats these series of operator actions as independent events in quantifying the operator failure probabilities. This appears to be too conservative an assumption since operators are trained and emergency operating procedures are written in such a way that these actions are an integral operation for a single function and thus they become highly dependent events.

The loss-of-offsite power initiator (T_1) leads to dominant sequences $T_1L_1D_1F$ and $T_1D_3WD_1F$ that are characterized by battery depletion and by pump seal LOCA, respectively. $T_1L_1D_1F$ represents a sequence initiated by loss of offsite power and followed by failures of onsite ac power and AFWS trains (L_1) including the turbine-driven pump due to battery depletion after 4 hours. High-pressure injection (D_1) and containment spray recirculation (F) also fail in this SBO sequence. $T_1D_3WD_1F$ is a sequence in which reactor coolant pump (RCP) seal LOCA occurs due to failures of seal injection flow (D_3) and RCP thermal barriers (W) resultingfrom loss of offsite power and failure of onsite ac power. Containment-spray recirculation (F) also fails in this sequence. T_1 is not assessed as important in IDCOR. This is partially because of the fact that the RCP seal LOCA scenario associated with T_1 is not considered in IDCOR. Another factor may be the difference in the recovery model related to the battery depletion time and ac power recovery time assumed in the two studies.

The ASEP study (in particular, Ref. 2) also identified loss of a dc bus as a significant contributor. This leads to several sequences as listed in Table A.6. We noted that there is a striking dissimilarity between the contribution of dc Bus I and Bus II for some of these sequences (e.g., CDFs for $T_{DCI}D_3WD_1H_3$ and $T_{DCII}D_3WD_1H_3$ are 2.3E-7 and 1.0E-5, respectively). It appears that this strong dependence on DC Bus II failures is because of an arbitrary assumption that Train A of the two charging pumps is normally operating. Unless there is other physical and configurational differences between the two trains of the systems involved in the sequence, the results of arbitrary modeling assumptions such as this carry a misleading message that dc Bus II is more important than dc Bus I.

The IDCOR study did not identify T_{CCW} and T_{DCX}. This is, as mentioned before, appears to be because of the limitation of grouping all transients other than loss-of-offsite power into one initiator at the beginning of the analysis instead of considering system dependent failures for specific initiating events before grouping.

The difference between IDCOR and ASEP in the estimate of CDF due to intermediate LOCAs (S_1) is attributed to the RCS "depressurization" (secondary blowdown) operation and the "core vulnerability" treatment. The IDCOR baseline analysis and ASEP study did not give credit to the RCS depressurizaiton operation (which depressurizes the reactor coolant system by blowing down the steam generators and uses the low-pressure injection system (LPIS)) in case of high-pressure injection systems HPIS) failure. The IDCOR committed risk pro-

file incorporated this operation in the analysis. The ASEP study identified S_1F and S_1G_1 sequences as core vulnerable sequences and then a fraction (0.13) of their frequencies was assumed to lead to core damage, while the IDCOR studies assumed S_1F and S_1G_1 as core damage sequences from the beginning. Here F and G_1 stand for containment spray recirculation and containment heat removal with containment spray recirculation trains, respectively.

Both studies estimated similar frequencies for the interfacing systems LOCA (V sequence) which deserves special attention because of its potentially high consequence. The IDCOR study reduced the RSSMAP estimate 4.6×10^{-6} per year to 9.0×10^{-7} per year giving credit to yearly testing and surveillance planned for the check valves between the LPIS and the RHR system. The ASEP study estimates 3.3×10^{-7}/year in Ref. 3, giving credit to a potential operator recovery action to isolate the interfacing LOCA by remotely closing the appropriate isolation valve in the low pressure line.

Neither study considered the steam generator tube rupture event. Although this initiator may not contribute significantly to the overall core damage (Ref. 15 indicates that this initiator contributes about 2.5% to the overall CDF in the Midland Nuclear Plant), it could contribute significantly to early fatalities. This may occur because of an open pathway in the secondary side at the time of core damage, resulting in a direct release from the primary side to the atmosphere through the ruptured steam generator.

A.2 Core-Meltdown Phenomena and Containment Response

This section contains a list of available Sequoyah Plant accident analyses (Section A.2.1), a comparison of available results from IDCOR and SARRP analyses for containment response, radioactivity release and consequences (Section A.2.2). The discussion of additional accidents analyzed by NRC contractors for this plant is in Section A.2.3. The differences between IDCOR and SARRP analyses identified in Section A.2.2 are discussed in Section A.2.4. Model differences and important emergency system features, which could diminish the consequences of analyzed accidents, are also discussed.

A.2.1 Containment Performance

The Sequoyah Nuclear Plant Accidents were analyzed by Sandia National Laboratories (SNL) in the RSSMAP study,[4] by Battelle Columbus Laboratories (BCL) in BMI-2104[9] and a recent BCL study[10] in support of NUREG-1150. The industry Sequoyah study is reported in Volume 23.1S[11] and Volume 21.1[1] of the IDCOR report.

Table A.9 gives the time of containment failure for each analysis. This time defines the release time of radioactivity which in turn affects the consequences of such release.

The results presented in Table A.9 were based on three different assumed pressure limits for containment failure (30 psia, 60 psia, and 65 psia) and are therefore, difficult to compare directly. This pressure limit was changed after the Reactor Safety Study (RSS) in accordance with additional analysis of the Sequoyah containment structure. This was one of the reasons that previous studies[4] predicted containment failure earlier than in the more recent studies.[1,10,11]

In comparing the IDCOR studies with SARRP studies one can observe two general trends: IDCOR predicts no containment failures because of hydrogen burns and all the IDCOR predicted containment failures are delayed in comparison to the SARRP results. In addition, the number of accident sequences analyzed by the NRC contractors is substantially larger than the sequences analyzed by IDCOR.

The containment response for the cases analyzed by IDCOR in Table A.9 and similar cases analyzed for SARRP are given in Tables A.10 through A.13.

S_2D Accident - (Failure of ECC Injection System, D)

The comparison of S_2D accident in Table A.10 shows very similar timing for core uncovery, core melt and RPV failure. However, the time of ice depletion is much longer in the IDCOR analysis than in the RSSMAP analysis. The pressure in containment rises to 30 psia in about 12 hours, when the containment fails in the RSSMAP analysis. In the IDCOR analysis, the pressure never rises above 21.5 psia (at 2.81 hours) and the pressure drops thereafter. This is demonstrated on the lower containment pressure of S_2D IDCOR calculations, shown in the Figure A.5. In this figure the solid line corresponds to two spray pumps operating (assumed in the IDCOR base case) and the dotted line corresponds to one spray pump operating (IDCOR minimum safeguard case). The RSSMAP calculations assumed that only one spray pump operates. Figure A.5 indicates that after 7.5 hours the IDCOR spray cooling system removes more heat than is generated by core decay heat (both the curves in Figure A.5 start dropping after 7.5 hours) and the containment is predicted to never fail. The steep rise of pressure around 5 hours (Figure A.5) coincides with ice depletion (see Table A.10, S_2D of IDCOR).

The delayed ice depletion and pressure drop after 7.5 hours for the IDCOR case, compared to RSSMAP analysis indicates that IDCOR cooling is more effective than the cooling modeled in RSSMAP, even with only one spray pump operating. (There are also differences in noncondensable gas generation rates, see Section A.3.2).

S_2H Accident - (Failure of ECC Recirculation System, H)

For recirculation failure the comparison in Table A.10 shows faster core uncovery, melting and RPV failure in the RSSMAP[4] calculations, than in the IDCOR[11] calculations. This indicates again, better cooling of the core in the IDCOR case with sprays operating. The pressure rises in the RSSMAP calculations to 30.0 psia at 13.3 hours, when the containment is assumed to fail. In the IDCOR case, the pressure rises to 21.4 psia and later drops without containment failure as shown in Figure A.6. The fifth column in Table A.10 shows the containment failure at the time of RPV failure, (1.83 hours) because of hydrogen burning.

S_2HF Accident (Failure of ECC and Sprays Recirculating Systems)

The comparison in Table A.11 for all 6 cases, shows somewhat faster core uncovery and core melting in the IDCOR[11] calculations then in the RSSMAP calculations. However, the RPV failure is at about the same time for both IDCOR and RSSMAP results (3.3 to 3.2 hours) for the "drains open" case.

The 30.0 psia pressure in containment is reached at 3.28 hours (RSSMAP - drains open) or 5.02 hours (RSSMAP - drains blocked), which is assumed to cause containment failure. The same pressure is reached in the IDCOR case later (5.2 hours) for open drains. The IDCOR containment failure pressure (65.0 psia) is reached at 9.54 hours (drains open). But, the IDCOR results never predicted containment failure because of hydrogen combustion (γ - case for S_2HF). For comparison, Table A.11 shows the later BCL calculations with hydrogen combustion (γ - cases) calculated in 1984[9] and 1986.[10] The S_2HF-γ case from 1984[9] shows a somewhat delayed RPV failure (at 4.33 hours) but followed at 5.87 hours by hydrogen combustion which raises the pressure to 121 psia and fails the containment. The case S_3HF-γ from BCL,[10] presented in Table A.11, is the case of pump seal failure, causing a leak of 0.78 inch diameter in the primary system (compared to a 2.0 inch diameter leak in the S_2HF case). In the S_3HF with drains open the core uncovary, core melt and RPV failure is delayed. The RPV fails at 6.84 hours followed closely by a hydrogen burn at 6.87 hours, which causes containment failure.

The pressure histories in the containment for S_2HF-γ BCL[9] and S_3HF_1-γ BCL[10] are given in Figures A.8 and A.9. Both of these figures show very fast decompression of the containment, after the hydrogen burn. This rapid decompression is caused by the large failure area (7 ft^2) assumed by BCL. The IDCOR analyses assume only a 0.10 ft^2 opening after containment failure, which leads to a slow containment decompression (see Figure A.7).

Note that in Table A.11, for the IDCOR analyses, the fraction of reacted cladding and released hydrogen is about half of the hydrogen generation predicted by BCL. Thus for IDCOR, no large hydrogen burns are predicted to occur.

TMLB' Accident

The comparison in Table A.12 shows rather similar times for core uncovery, core melt, and RPV failure for the IDCOR and BCL[9] cases. For the RSSMAP[4] calculations the core uncovery and core melt is delayed, compared to IDCOR and the RPV failure time is slightly longer. However, the pressure in containment rises faster in the BCL[9] calculations than in the IDCOR results, as can be seen in Figures A.10 and A.11. The BCL pressure rises to 60.0 psia (assumed failure pressure) at 9.21 hours, but for the IDCOR[11] case the containment is predicted to fail in 27 hours. The decompression of containment is very rapid in the BCL calculations (see Fig. A.10) and very slow for IDCOR (see Figure A.11), caused by differences in the assumed containment opening (7 ft^2 or 0.12 ft^2, respectively.). For the hydrogen burn case, TMLB'-γ, from BCL[9] the containment pressure response is shown in Figure A.12. This case shows a large pressure peak at 2.03 hours when the RPV fails causing containment failure). A number of smaller peaks occur in the failed containment because of subsequent hydrogen burns. In the RSSMAP[4] calculations, the containment fails at 5.47 hours (at 30 psia).

$T_{23}ML$ Accident

The IDCOR and BCL timing results are similar for core uncovery, melt and RPV failure. However, the containment pressure response in BCL is very different from the IDCOR calculations (see Figs. A.13 and A.14). In the IDCOR case the pressure rises to a peak of 23 psia at 2.91 hours (pressure vessel

failure). It rises again after ice depletion (5.84 hours) and after 8 hours it starts decreasing when the containment atmosphere cooling exceeds the decay power of the debris. In the BCL TML-δ case[10] the containment pressure rises to 52 psia at RPV failure (2.63 hours), because of hydrogen burning. After this peak the pressure in containment slowly rises (with several smaller hydrogen burn peaks). until 13 hours (end of calculations). In the TML-δ BCL calculation, cooling of the containment atmosphere never exceeded the decay power of the core debris. The containment did not fail in either the IDCOR nor the BCL calculations (but the pressure calculated by BCL is continuing to rise and eventual containment failure is expected).

The BCL prediction[9] for TML-γ gives the containment pressure response shown in Figure A.15. There is a large pressure peak because of a hydrogen burn at 2.62 hours, which also causes failure of containment. The RSSMAP-δ[4] case has assumed containment failure at 30 psia which was reached at 11 hours.

AD Accident (Failure of ECC Injection System, Intermediate to Large LOCA)

The comparison in Table A.12 shows much faster core uncovery and core melt in the RSSMAP[4] analysis then in the IDCOR analysis. The difference appears to be because of the assumed LOCA size (large in RSSMAP[4] and intermediate in response predicted by IDCOR is given in Figure A.16. This response is similar to Figure A.5 for the S_2D accident. The pressure jumps to a peak of 22.5 psia at reactor vessel failure (1.1 hours) and after ice depletion (3.15 hours) rises again above 20 psia. However, because of operation of the containment sprays after 6 hours, the pressure decreases continuously, when decay power is surpassed by the cooling power of sprays. The "minimal safeguard case," marked by the dotted lines in Figure A.15, uses only one spray pump operating, and it behaves the same way as the base case, except that the maximum pressure is slightly higher (23.5 psia). RSSMAP[4] predicts the assumed containment failure pressure (30 psia) will be reached at 6 hours.

V, β and TMLU-SGTR

The comparison of some available data for V and β (interfacing LOCA and containment isolation failure) sequences are given in Table A.13. The V sequence is predicted to result in very late RPV failure (20 hours) by IDCOR and rapid RPV failure (1.52 hours) by RSSMAP.

The IDCOR β/W and S_2H-β sequences have faster RPV failure - at 7 hours, compared to 20 hours for the V accident.

The TMLU-SGTR accident was treated by BCL.[10] In this accident core melt starts at 2.12 hours and core slump begins at 2.57 hours, with RPV failure predicted to occur at 2.82 hours. In this accident it is assumed, that the core slumps at 2.57 hours causing the steam generator tube rupture (SGTR). The steam generator tube failure is assumed to allow radioactive aerosols to escape into the environment via the steam generator secondary relief valve. This release lasts only until the RPV fails (at 2.82 hours).

A.3 Comparison of Fission-Product Releases

The comparison of the release of radioactivity from the IDCOR and SARP calculations for δ failure mode is given in Table A.15. From this table, one can make several observations:

(1) The IDCOR calculations from the risk reduction report[1] and the Sequoyah containment analysis[11] give inconsistent results for the same case: Reference 11 data gives about 1/10 the releases of Ref. 1 data (for S_2HF and TMLB' accidents).

(2) The IDCOR[11] releases for TMLB' are about the same for CsI and CsOH (with the Te release about 100 times larger in the BCL[9] analyses, than in IDCOR[11] analyses). This seems to indicate similar modeling of aerosol deposition in IDCOR[11] and BCL.[9] The TML and AD releases predicted by IDCOR is zero since there is no containment failure. The BCL TML and AD releases are also almost zero (only design leakage from the closed containment is assumed for the TML-δ case).

(3) All the S_2HF-δ IDCOR cases are compared with RSSMAP. They show the IDCOR[11] releases to be only about 1/1000 to 1/10000 of the RSSMAP releases.

Other fission-product release results for γ mode failure (hydrogen burns) and for containment bypass and isolation failure (V and β cases, respectively) are compared in Table A.15. The releases for S_2HF-γ from BCL[9] and S_3HF-γ by BCL[10] are similar for CsI or Cs, but 10 to 20 times smaller for Te, Sr in Ref. 10. The S_2 case assumes a 2 inch LOCA, the S_3 case assumes 0.75 inch equivalent hole in the primary pump seals.

The V sequence (containment bypass) release for RSSMAP[4] is about equal to the RSS PWR2 category release. In the case of V releases in IDCOR[1] the released fraction is extremely low, much lower than the β/W case, which also has an open containment.

A.3.1 Analysis of Sequoyah Plant Accidents Not Treated in IDCOR Studies

A number of other accidents were analyzed in the SARP sdudies for the Sequoyah plant. The summary of all March results from Reference 4 are given in Table A.16 and the summary of the CORRAL results for fission-product deposition are given in Table A.17. The S_2H, S2HF, TMLB' and TML accidents from those two tables were discussed previously in Section A.2.1 and A.2.2, in comparison with IDCOR analyses for similar accidents.

The new results in these tables are for AD-α, AD-γ, S_2HF-α, AHF accidents and variations of TMLB' and V accidents. The results in Table A.17 are listed for all the accidents analyzed in RSSMAP.

The recent results of BCL[9,10] generally show, for similar accidents, lower releases then given by RSSMAP (Table A.18). The reason for this difference is, more detailed modeling of fission-product transport and deposition in the BCL[9,10] studies, i.e. in the primary system (using CORSOR-TRAPMELT-MERGE codes) and in the containment (using VANESA, ICEDF codes).

A.3.2 IDCOR and SARP Input and Modeling Differences

Figures A.5 to A.16 demonstrated very different containment response for the BCL and IDCOR analyses. Some of these differences can be explained by different input assumptions. A comparison of input data is given in Table A.18. There are a number of significant differences in the input data. The similarities and differences are discussed below:

(1) Reactor power is smaller in the IDCOR analysis (95.5% of the BCL power) but also the ice content is smaller (85.5% of the BCL analysis) and the ice temperature is somewhat lower (by 50F). These effects give a similar ratio of power to ice cooling capacity and do not account for the major differences in the results.

(2) The masses of UO_2 are similar: The Zr mass is about 6% higher in the BCL analysis.

(3) The containment volumes are similar but the temperature of the upper containment and all material in it is lower by 15F than in the BCL analysis.

(4) Concrete and steel volumes (calculated from different items of BCL and IDCOR data) are similar. However, the IDCOR volumetric heat capacity of concrete is smaller.

(5) There are significant differences in the engineering safety systems (ESS): The LPI pumps and the containment sprays have twice the flow rate in the IDCOR analyses (2 pumps and sprays are assumed to be operating) than in the BCL analyses. This provides twice the cooling rate.

    The heat exchanger flows in recirculation and the containment spray heat exchangers are two to three times larger (in the primary circuits). This provides a cooling capacity of 2x152% for ECC heat exchangers and 2x100% for spray heat exchangers. As a result the core cooling water and the containment are cooled two to three times better in the IDCOR calculations than in the BCL calculations.

    Finally, the UHI tank and accumulator water is 22% and 13% larger in the IDCOR case, than in the BCL case.

    The results of these IDCOR cooling assumptions is that the core and containment are cooled substantially better in the IDCOR analyses. This results in substantially lower pressure inside containment for the IDCOR results as observed in Figures A.5, A.6, A.13, A.16 or in a delay of the pressure rise as observed in Figures A.7 and A.11.

(6) The IDCOR containment break area is very small compared to BCL (0.1 or 0.02 ft^2 compared to 7 ft^2 for BCL). As a result, in all the BCL cases the containment depressurizes rapidly after containment failure (see Figures A.8 through A.10, A.12, A.14, and A.15), but in the IDCOR calculations depressurization is very slow (see Figures A.7 and A.11).

In addition to the input differences, there are a number of differences between the IDCOR and BCL computer models. These differences are listed in the form of issues, which have been discussed by the NRC and IDCOR staff in a series of meetings. These issues are listed in Table A.19. Out of the 18 issues, a subset of 11 have been identified that are appropriate to the subject of this section. Each issue is briefly discussed in the following sections. Differences between the IDCOR and BCL analyses will be identified and their significance indicated.

In-Vessel Hydrogen Generation (NRC/IDCOR Issue 5)

There are significant differences between the IDCOR and BCL predictions of hydrogen (H_2) generation during in-vessel core melting. During the early stages of core heatup and degradation (while the fuel rods are still in place in the core region), both IDCOR and BCL predict similar H_2 generation. However, after the fuel rods and cladding begin to melt and relocate in the bottom ofthe reactor vessel, the BCL analysis indicates, substantially, more H_2 generation than the IDCOR analysis. Often twice as much H_2 in the BCL analysis compared to IDCOR. This difference is indicated in Tables A.11 and A.12, in the line "fraction of clad reacted" which are proportional to H_2 generation.

Hydrogen is important to containment loading because it is a combustible and noncondensable gas. The larger amount of H_2 generated in-vessel in the BCL and SARRP analyses leads to a higher predicted containment pressure prior to vessel failure than in the IDCOR analysis.

Core Slump, Core Collapse, and Reactor Vessel Failure (NRC/IDCOR Issue 6)

The IDCOR core slump model assumes that after 40% of the core has melted, it will relocate into the bottom of the reactor vessel, which will then rapidly fail because of local penetration failure. Thus, only a small fraction of the core will be initially released from the reactor vessel. The remainder of the core melts down over a much larger time period. On the other hand, the BCL core slump model assumes total collapse of the core into the bottom of the reactor vessel after 75% of the core is predicted to melt. Thus, most of the core debris is available to be released when the vessel is predicted to fail in the BCL model. This modeling tends to increase the hydrogen available for detonation or deflagration at vessel break (Issue 5), and the potential for direct heating (Issue 8) as well as promote a more rapid core-concrete attack (Issue 10) with more aerosol generation (Issue 6).

Direct Heating of Containment (Issue 8)

The pressure rise in containment because of direct heating is directly proportional to the quantity of core debris dispersed from the reactor vessel. The BCL analysis predicts significantly more debris release at vessel failure than the IDCOR analysis. Thus the potential for early containment failure because of direct heating is higher in the BCL analysis.

The assumption that all the core debris is released at vessel failure (BCL analysis) is clearly conservative and the SARRP[12] uncertainty study has addressed the possibility of small debris fractions. The IDCOR results appear to be too optimistic considering the lack of supporting large scale

experiments. However, spray operation may reduce the pressure pulse associated with direct heating by flooding the reactor cavity and would help quench the core debris. Containment sprays and cooling of sprayed water are thus very important to the timing and mode of containment failure.

Ex-Vessel Heat Transfer Model from Molten Core to Concrete (Issue 10)

Almost all cases analyzed by IDCOR assume that the reactor cavity is flooded by water from sprays, because of overflow via drains (drains open cases) from the upper reactor cavity into the lower reactor cavity. This mechanism allows hot debris to drop from the failed RPV into water and form a coolable deris bed under water. The solidified debris is assumed to stay solid without further attacking the concrete. In the BCL analyses the debris attacks concrete continually.

The IDCOR assumption appears to be overly optimistic given the lack of large scale debris coolability data. If the debris is not cooled, it will attack concrete and produce noncondensable gases, even if the top surface of the debris is covered by water. In view of the uncertainty, the slow pressurization of the containment resulting from the coolable debris assumption in the IDCOR analyses appears to be unjustified and the potential for the rapid containment pressurization in the BCL[11] analyses should be considered.

Containment Failure Because of Hydrogen Deflagration (Issue 17)

In the IDCOR analysis hydrogen deflagrations do not occur. This is because of the small amounts of hydrogen generated (about 1/2 of the BCL[11] predictions) and because of the presence of igniters that burn hydrogen continuously at low concentrations.

BCL predicts a large number of hydrogen deflagrations, which often cause containment failure (see all γ-cases in Table A.11 through A.15 and corresponding Figures A.8, A.9, A.12, A.14 and A.15). The SARRP uncertainty study[12] also indicates a substantial likelihood of containment failure because of hydrogen deflagration.

Containment Performance (Issue 15)

IDCOR assumes that a relatively small opening will occur which allows gradual leakage of the containment atmosphere to the auxiliary building. By comparison the BCL analysis assumes an opening large enough (7 ft^2) to rapidly depressurize the primary containment.

Fission Product Release Prior to Vessel Failure (Issue 1)

Both studies predict similar releases of the more volatile fission products during in-vessel core degradation with the exception of tellerium (Te). (IDCOR predicts about 1/10 the releases of BCL for Te for TMLB'-δ). However, in the IDCOR analysis, rubidium (Rb), zirconium (Zr), plutonium (Pu), lanthanum (La), barium (Ba), yttrium (Y), technetium (Tc), rhodium (Rh), and palladium (Pd) were omitted from the in-vessel release. This IDCOR omission will lead to somewhat smaller environmental doses than in the BCL cases. IDCOR has agreed to account for these omissions in future calculations.

Fission-Product and Aerosol Retention in the Primary System (Issue 4)

Differences in the initial primary system retention predicted by IDCOR and BCL are not too significant and differ by less than a factor of two. In the BCL modeling it is assumed that fission products retained in the primary system at the point of vessel failure are retained permanently. In the IDCOR analysis of the Peach Bottom plant, revaporization of these fission products after vessel failure was modeled.[4] This revaporization is not used in the IDCOR model for Sequoyah since the higher heat losses from the primary system are assumed to keep the Sequoyah primary system relatively cool.

Ex-Vessel Fission Product Release (Issue 9)

There are significant differences between the IDCOR and BCL analyses for fission product release as a result of core-concrete interactions. The higher releases of strontium (Sr) (and also La and Ce (cerium) groups) in Table A.15 in the SARP[9,10] analyses are because of the modeling of ex-vessel fission product release. Fission product release and inert aerosol generation during core-concrete interactions was not modeled in the IDCOR analysis. IDCOR argued that even for a dry cavity the aerosol generation during core-concrete interactions would increase aerosol density in containment and increase aerosol agglomeration or settling, thus reducing the predicted environmental release fractions relative to those predicted without this additional aerosol source. This IDCOR argument does not appear to be adequately supported. In addition, the IDCOR predicted core debris temperatures during core-concrete interactions are very high. Based on experimental evidence, the release of some of the refractory fission product groups would be expected at these elevated temperatures. The BCL and SARRP analyses currently model the release of the refractory fission products and the inert aerosols and the environmental release fractions are significant (refer to Table A.15).

Retention of Fission Products in Ice Beds (Issue 13) and Deposition Model in Containment (Issue 12)

The retention of fission products in ice beds and fission product surface deposition was modeled for BCL by using the NAUA and ICEDF codes. IDCOR used the MAAP-RETAIN code. The results are rather similar for IDCOR and BCL, as can be seen from Table A.15 for TMLB'-δ cases release fractions of CsI and CsOH (the only significant difference is in the Te release). RSSMAP modeled the aerosol deposition using only the CORAL code (without use of the TRAPMELT-MERGE module for fission product deposition in the primary system) and it produced about 1000 times larger releases than IDCOR and 100 to 10 times larger releases than BCL[9,10] (see comparison in Tables A.14 and A.15 for the same analyzed accidents).

A.4 Offsite Consequences

The consequence bins calculated by IDCOR are summarized in Table A.20. The radioactivity releases for this table are given in Tables A.11 to A.13 of the IDCOR Risk Reduction report.[1] The IDCOR results indicate that the ice-condenser containment is very effective in mitigating fission-product releases and the offsite consequences (Table A.20) are quite small. However, the SARRP[12] consequence results are very different than the IDCOR results for the ice condenser.

The biggest differences between SARRP and IDCOR appear to be containment performance and fission product releases during core-concrete interaction. The overall containment failure probabilities are shown in Figure A.17. SARRP estimates that there is about a 20% chance of early containment failure for Sequoyah core-damage sequences and nearly a 60% change of late containment failure. For late containment failure sequences the releases from core-concrete interactions tend to be important contributors to risk. IDCOR assessed the probability of early containment failure to be negligible. The SARRP results of the uncertainty study are depicted in Figures A.18 and A.19. The overall risk appears to be fairly consistent with the RSS. Although the MACCS calculation of latent-cancer fatalities is somewhat higher than the RSS because of new modeling of root uptake.[12]

A.5 Summary and Risk Insights

A.5.1 Core-Damage Profile

As has been observed in other PWRs, transients and small LOCA dominate the core-damage profile for the Sequoyah plant. (Note that the IDCOR study did not identify the SBO RCP seal LOCA sequences, the loss of component cooling water sequences, and the loss of a dc bus sequence.) It is noteworthy that the small LOCA (S_2) is the most dominant core-damage contributor both in the IDCOR study and in the ASEP study. Figure A.20 provides an illustration of the new sequence frequencies developed by ASEP.

Although there are a large number of contributing sequences, Table A.8 suggests that the CDF depends on only a few initiators. Thus, if the frequency of these initiators can be controlled, there is a reasonable expectation that the overall CDF will be controlled.

Although it is recognized that the qualitative accident sequence descriptors are rather general and broad and that similar hardware functional requirements and operator actions in the ice-condenser plants (Sequoyah, D.C. Cook, McGuire, Catawba, and Watts Bar) would lead to the same general accident sequences, a plant-specific examination (such as a failure mode and effects analysis coupled with a fault tree/event tree analysis) is needed in order to identify the plant specific vulnerabilities (e.g., in maintenance practices, operator training, and emergency operating procedures) to contribute to a given general sequence descriptor.

A.5.2 Consequence Analysis

The assessment of core-melt phenomena and containment response indicates that the ice-condenser containment provides a robust defense against early severe accident pressures and temperatures. However, differences in the IDCOR and SARRP assessments of containment response and fission-product release result in major differences in the predicted offsite consequences. The IDCOR assessment indicates that the containment will fail very late in the accident sequence (9 to 27 hours) if it fails at all. The SARRP results predict much earlier containment failure (3 to 20 hours) and much higher fission-product releases. Even with early containment failure the SARRP results predict substantial decontamination by the ice bed so that the fission-product releases tend to be lower than had been estimated by RSSMAP (based on Surry consequence bins).

A.6 References

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3. R. C. Bertucio et al., "Analysis of Core Damage Frequency From Internal Events: Sequoyah, Unit 1," Sandia National Laboratories, NUREG/CR-4550, Volume 2, February 1987.

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5. Sequoyah Nuclear Plant Final Safety Analysis Report, Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, April 1984.

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17. A. S. Benjamin et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Surry Power Station, Unit 1," Sandia National Laboratories, NUREG/CR-4551, in press.

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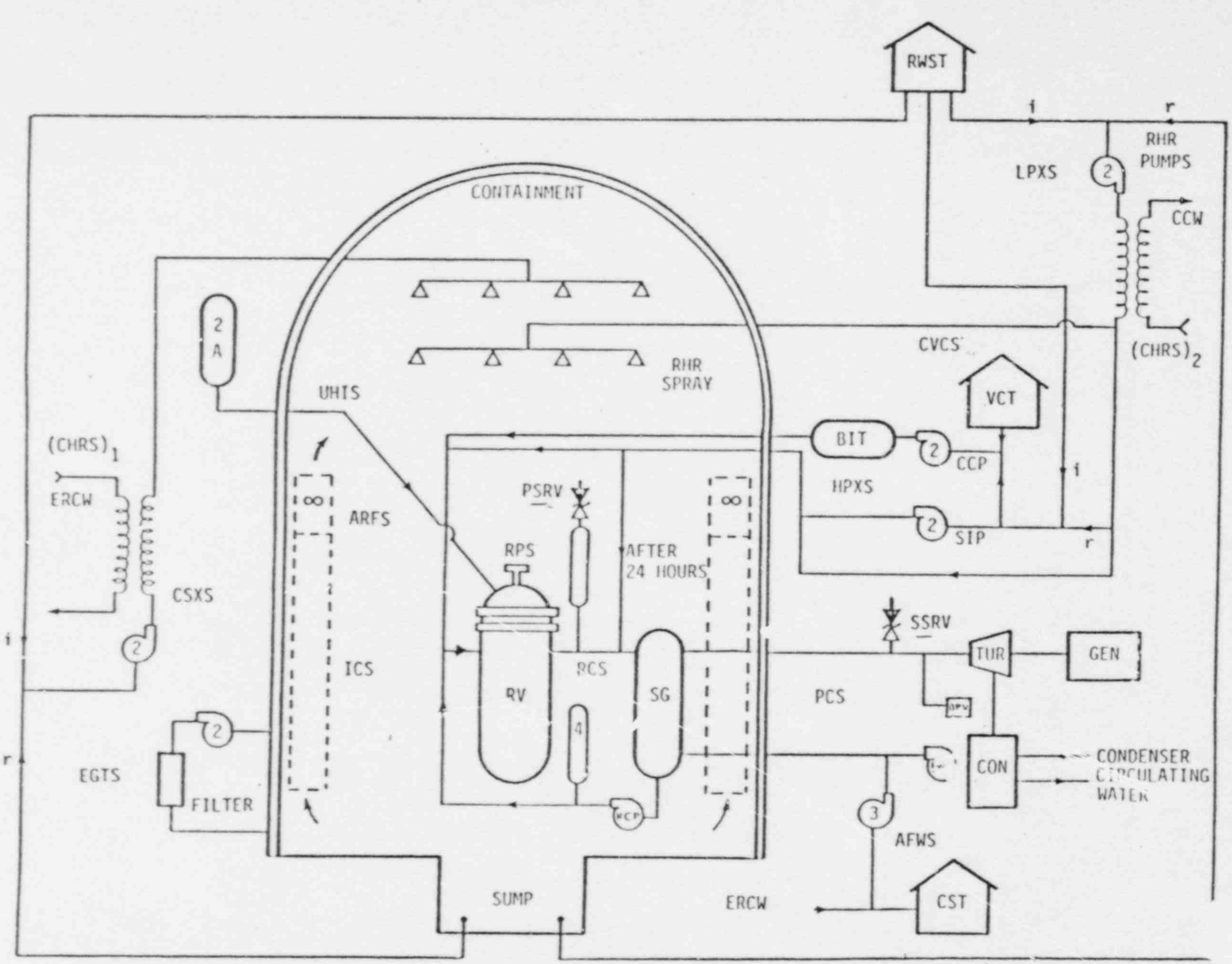


Figure A.1 A schematic of the Sequoyah engineered safety features (from Ref. 4).

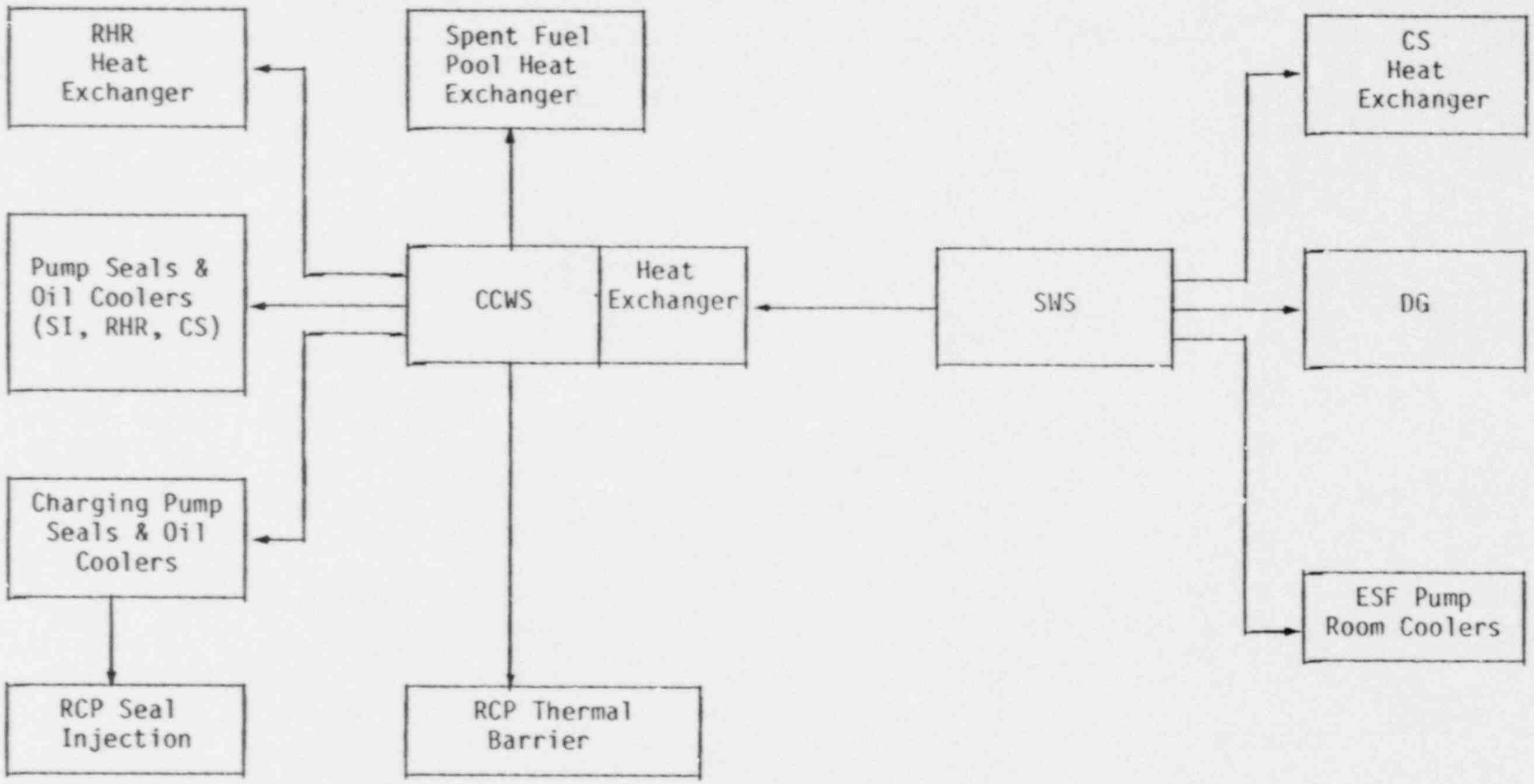


Figure A.2 A heat and fluid flow diagram for Sequoyah.

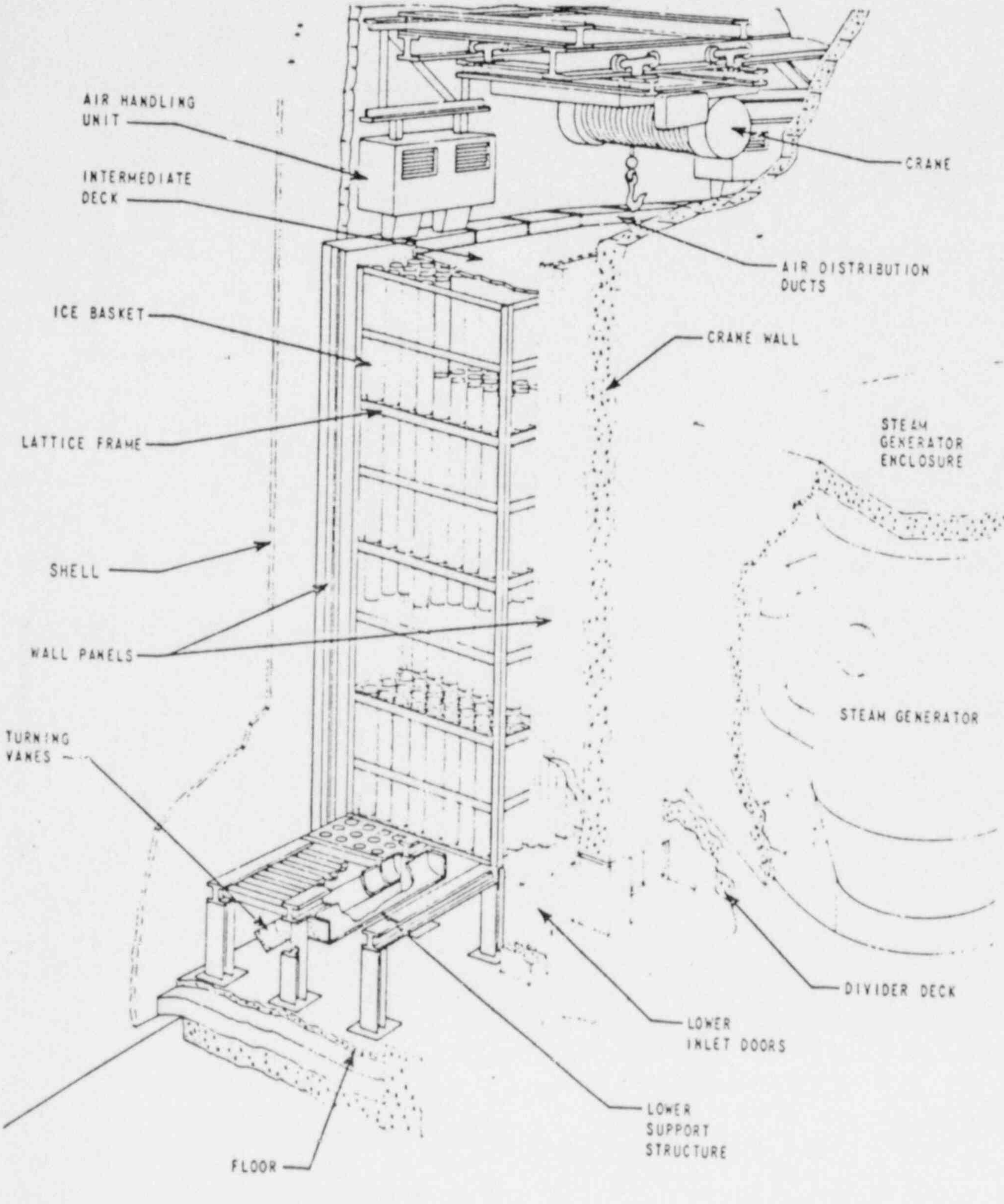


Figure A.3 A schematic of Sequoyah ICS (from Ref. 4).

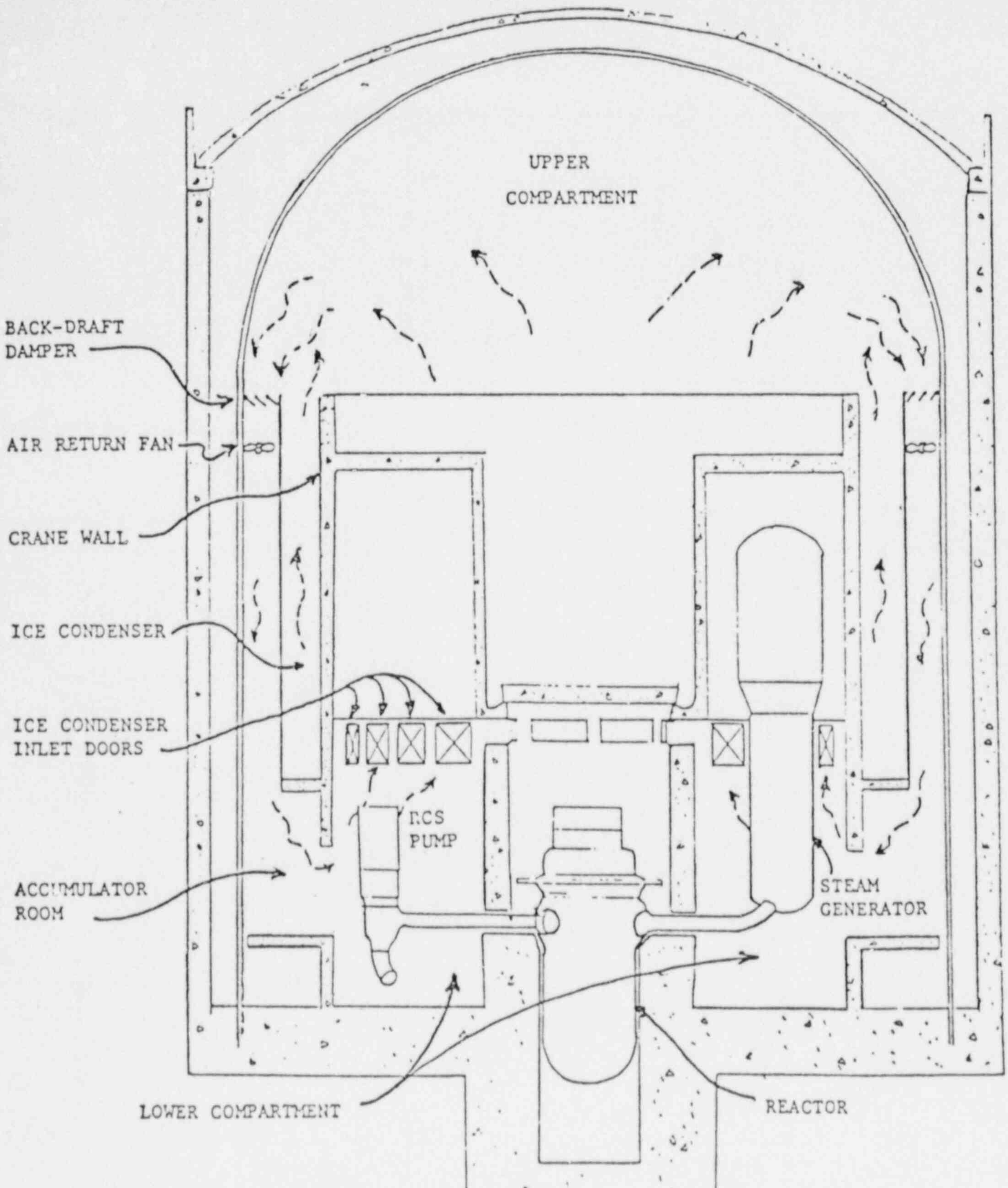
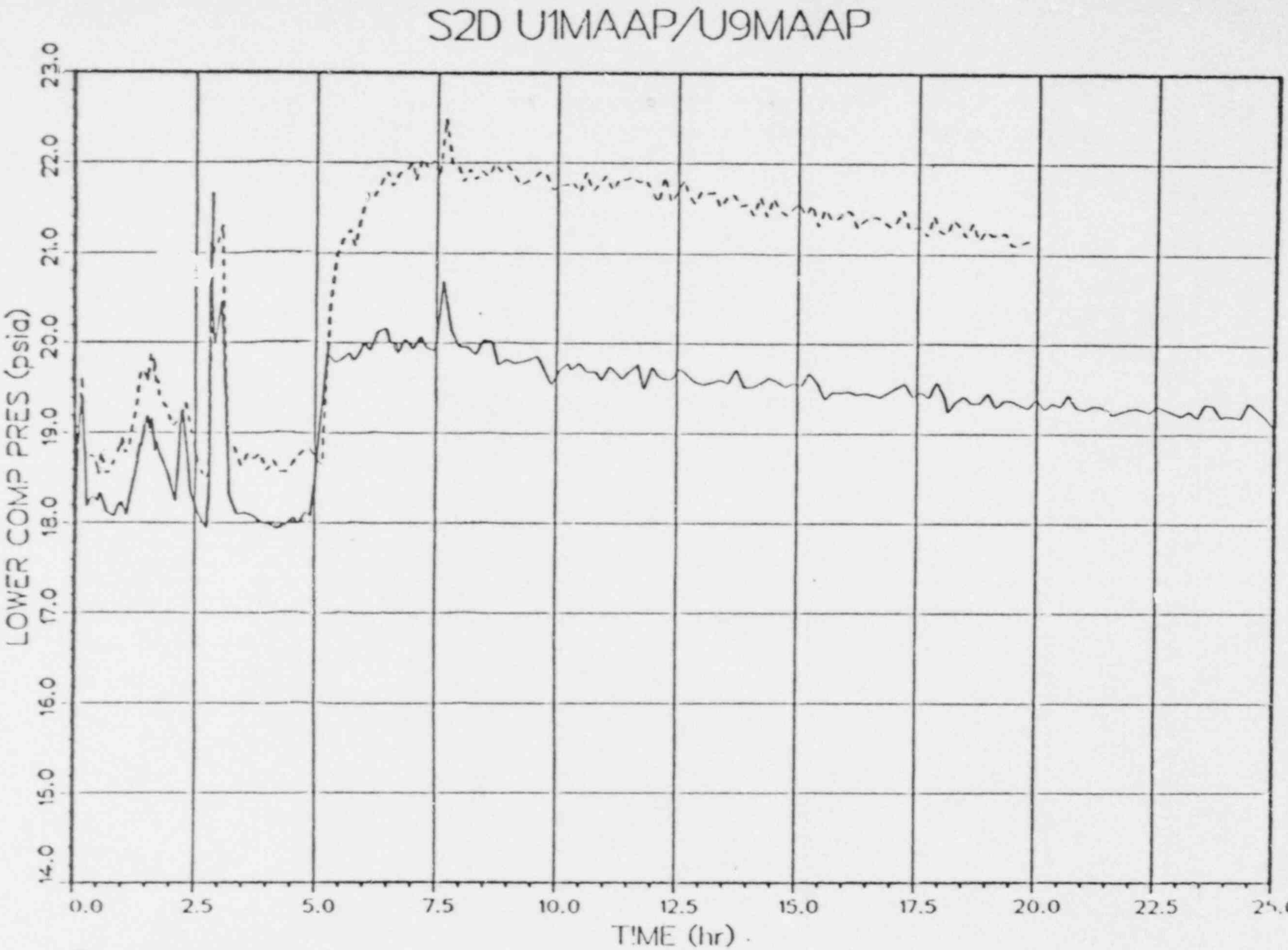


Figure A.4 A simplified flow diagram of Sequoyah ARFS (from Ref. 4).



——— Two spray pump operating
------ One spray pump operating

Figure A.5 Pressure in lower compartment for S_2D accident (IDCOR).[11]

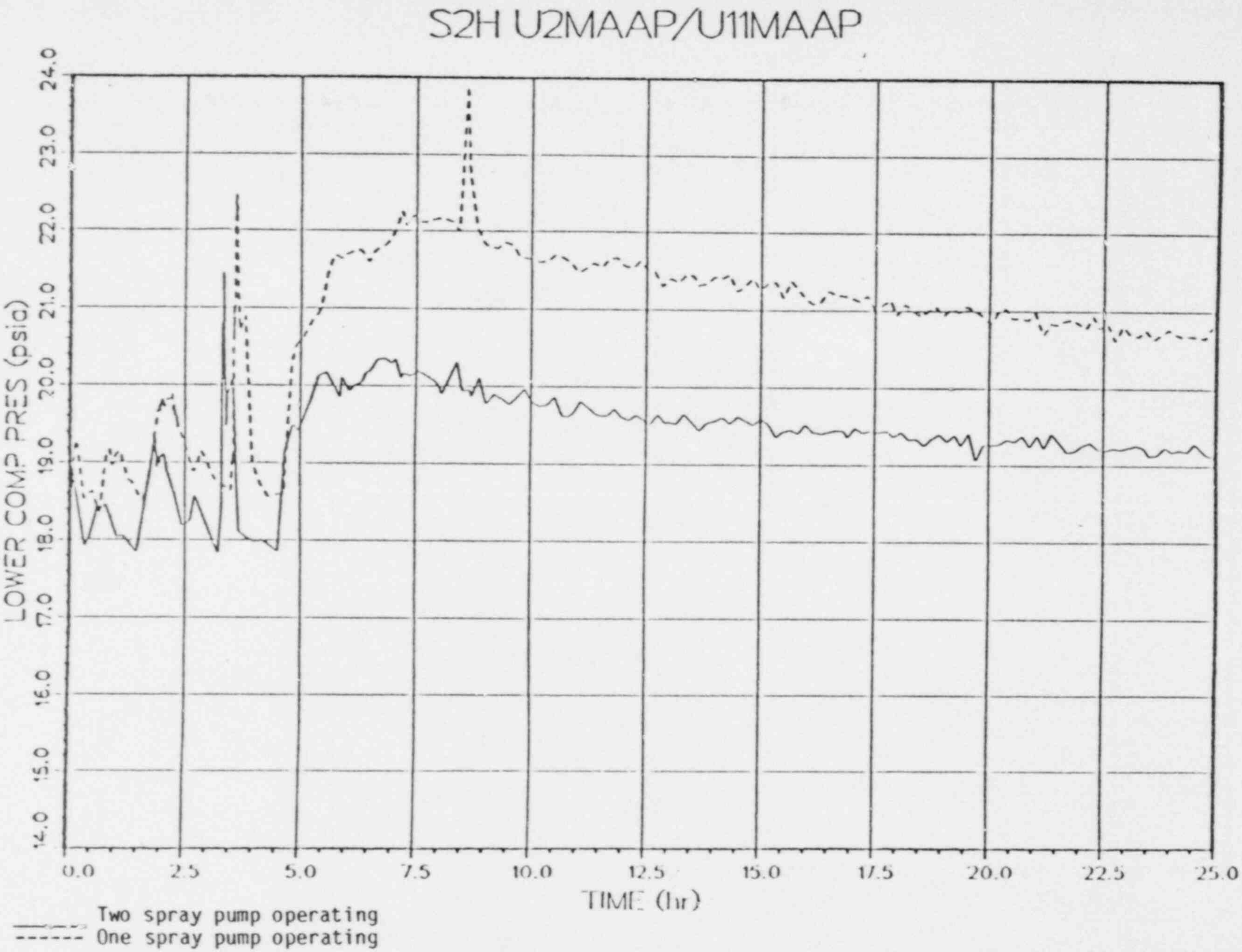


Figure A.6 Pressure in lower containment for S_2H accident (IDCOR).[11]

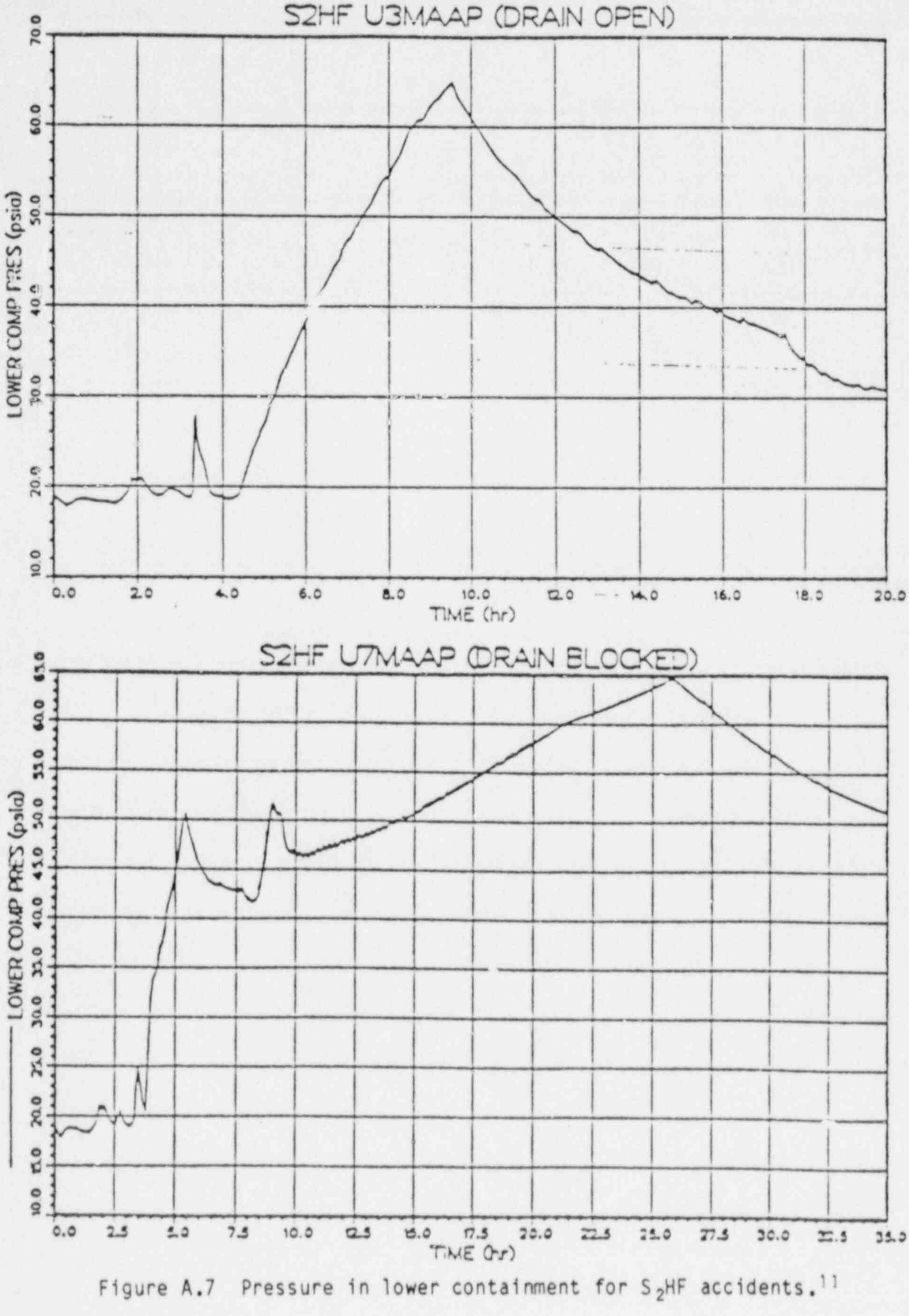


Figure A.7 Pressure in lower containment for S_2HF accidents.[11]

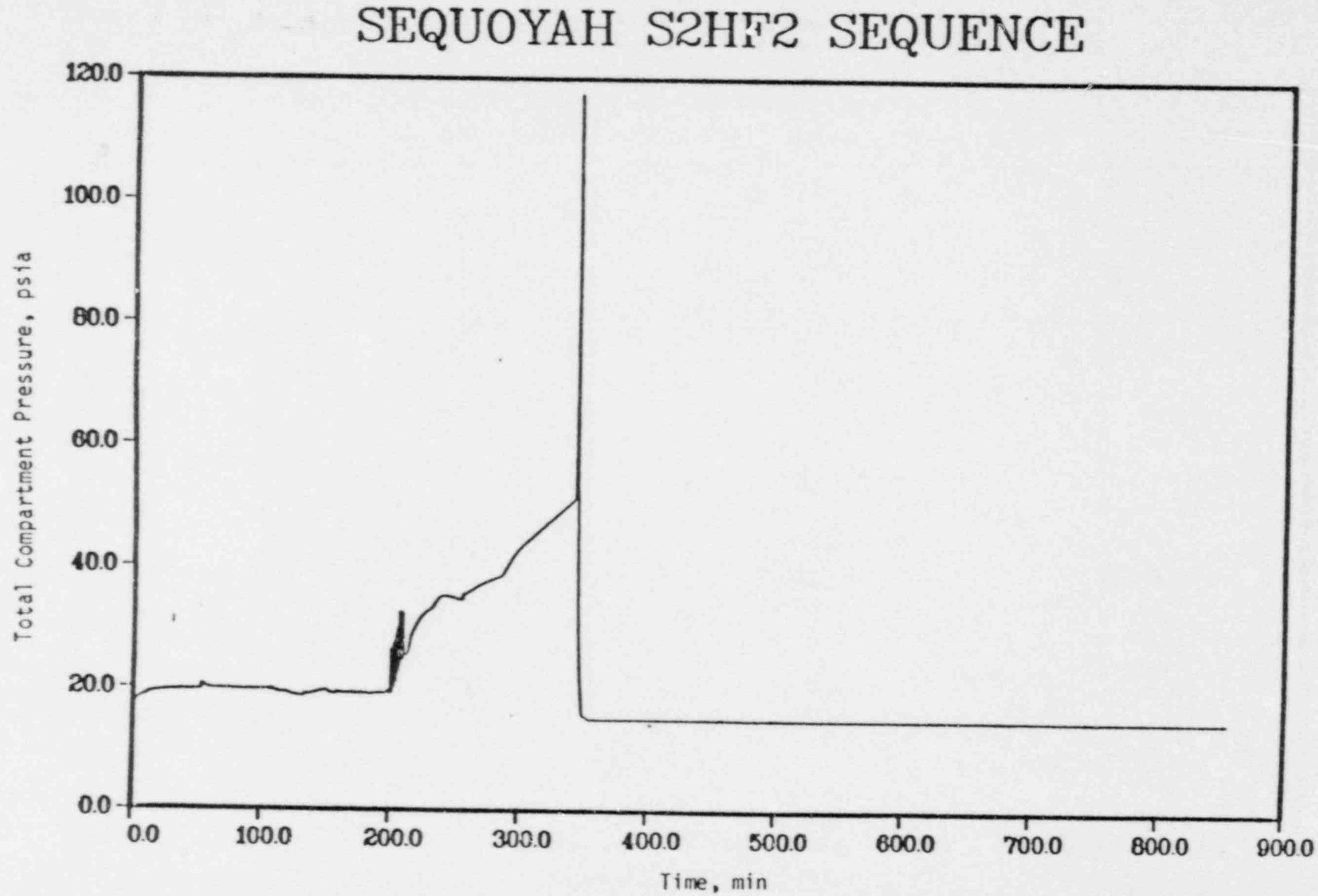


Figure A.8 Containment pressure response during Sequoyah $S_2HF-\gamma$ sequence.[9]

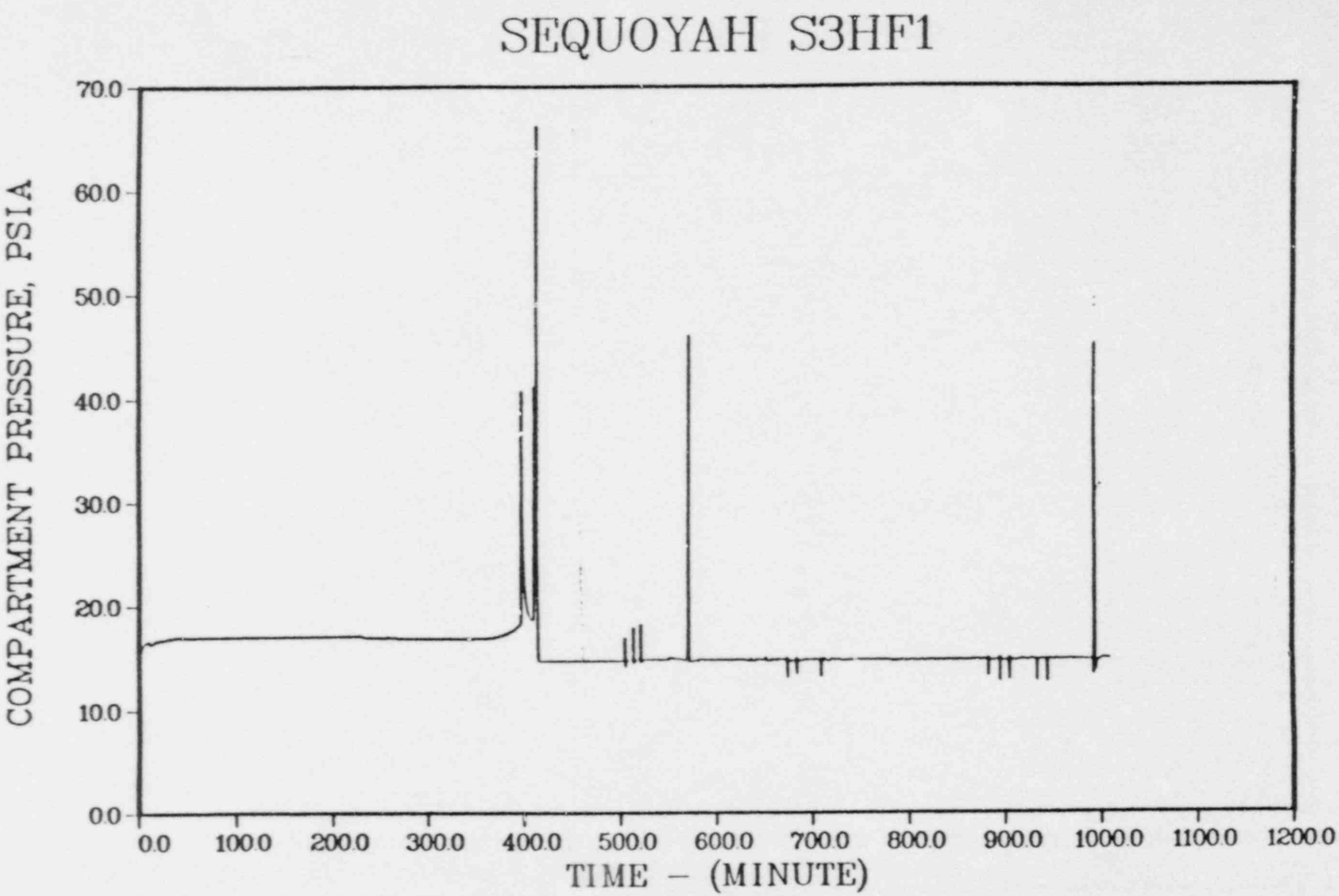


Figure A.9 Containment pressure response for S_3HF_1.[10]

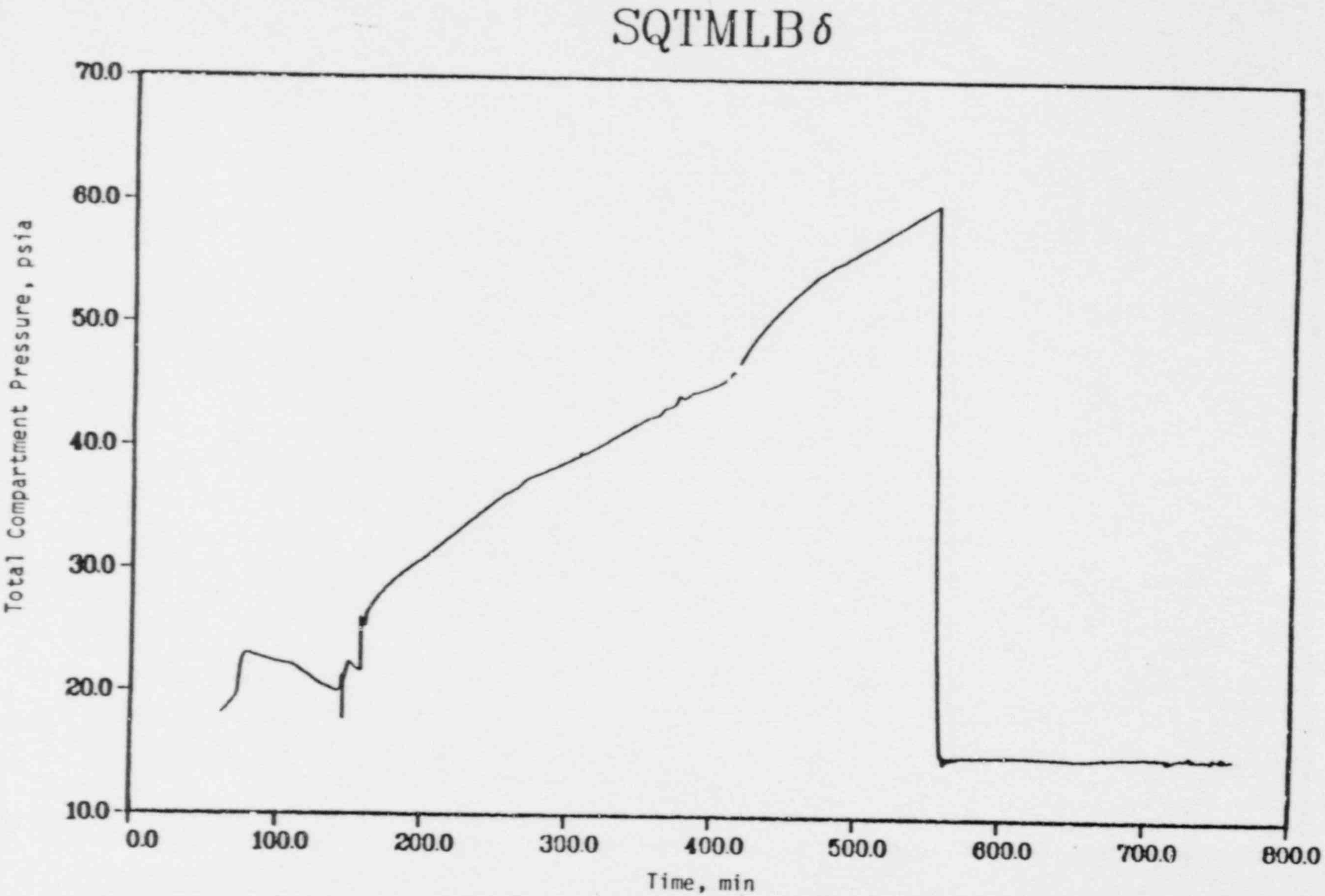


Figure A.10 Containment pressure response during Sequoyah TMLB'-δ sequence.[9]

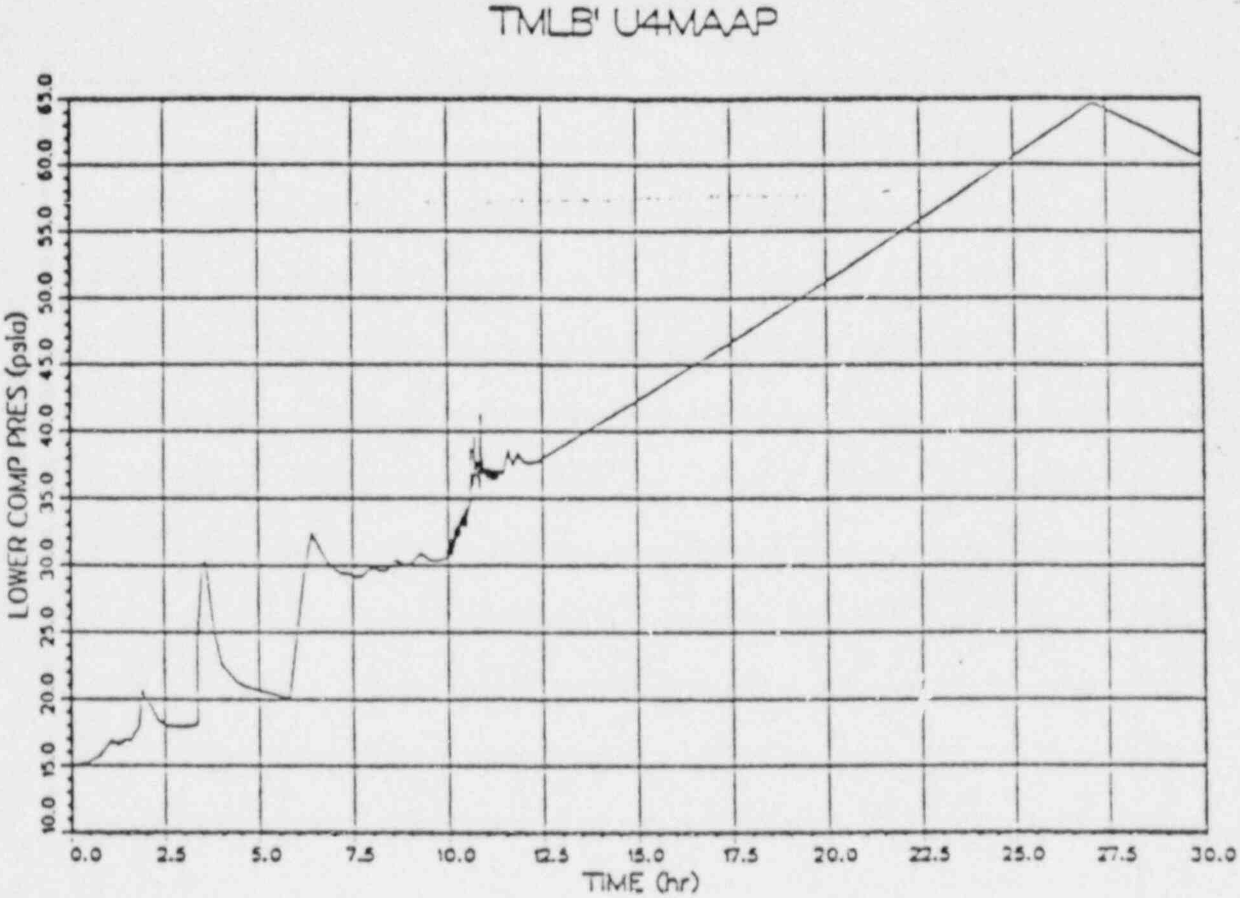


Figure A.11 Containment pressure response for TMLB'-δ accident, IDCOR.[11]

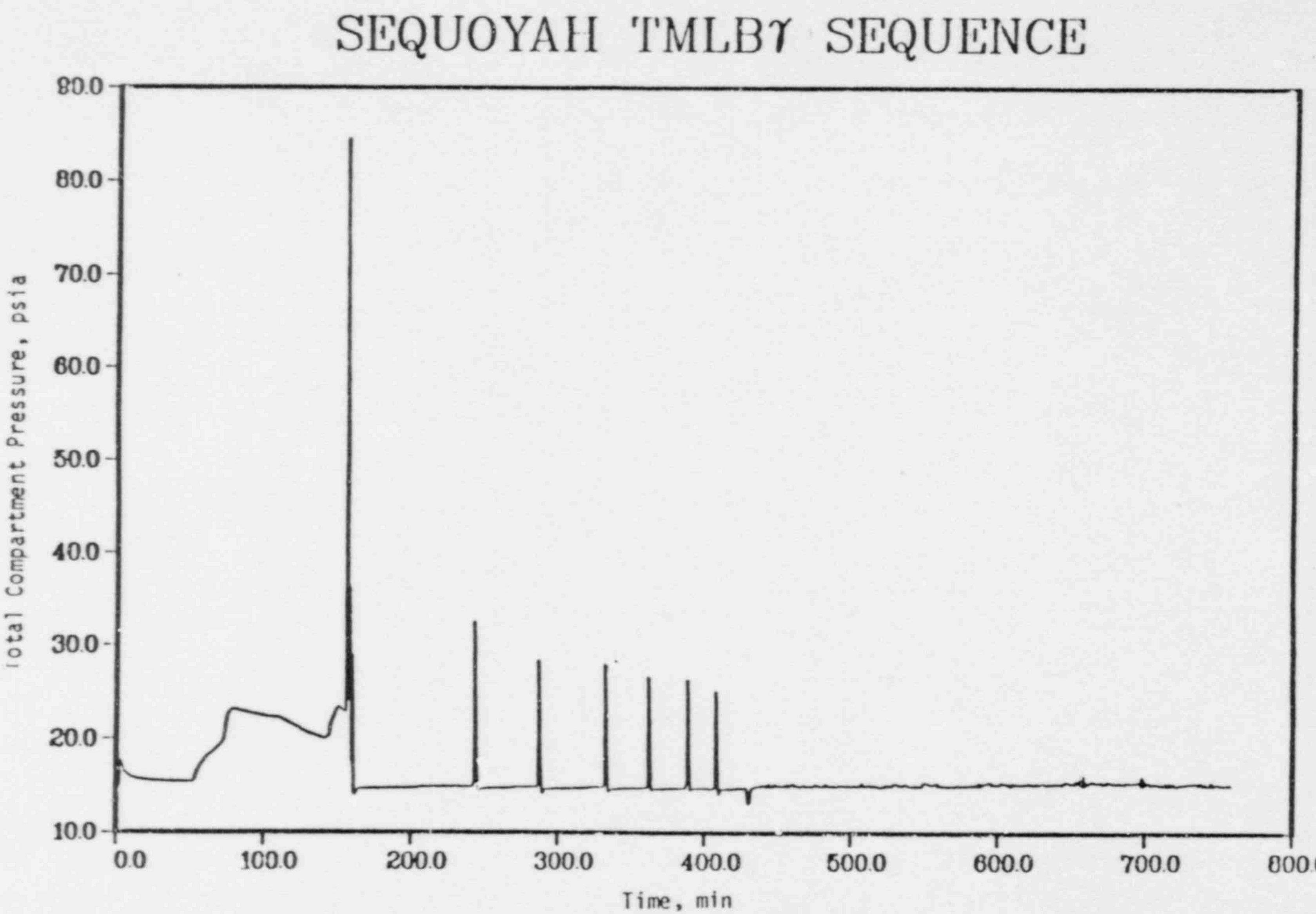


Figure A.12 Containment pressure response during Sequoyah TMLB'-γ sequence.[9]

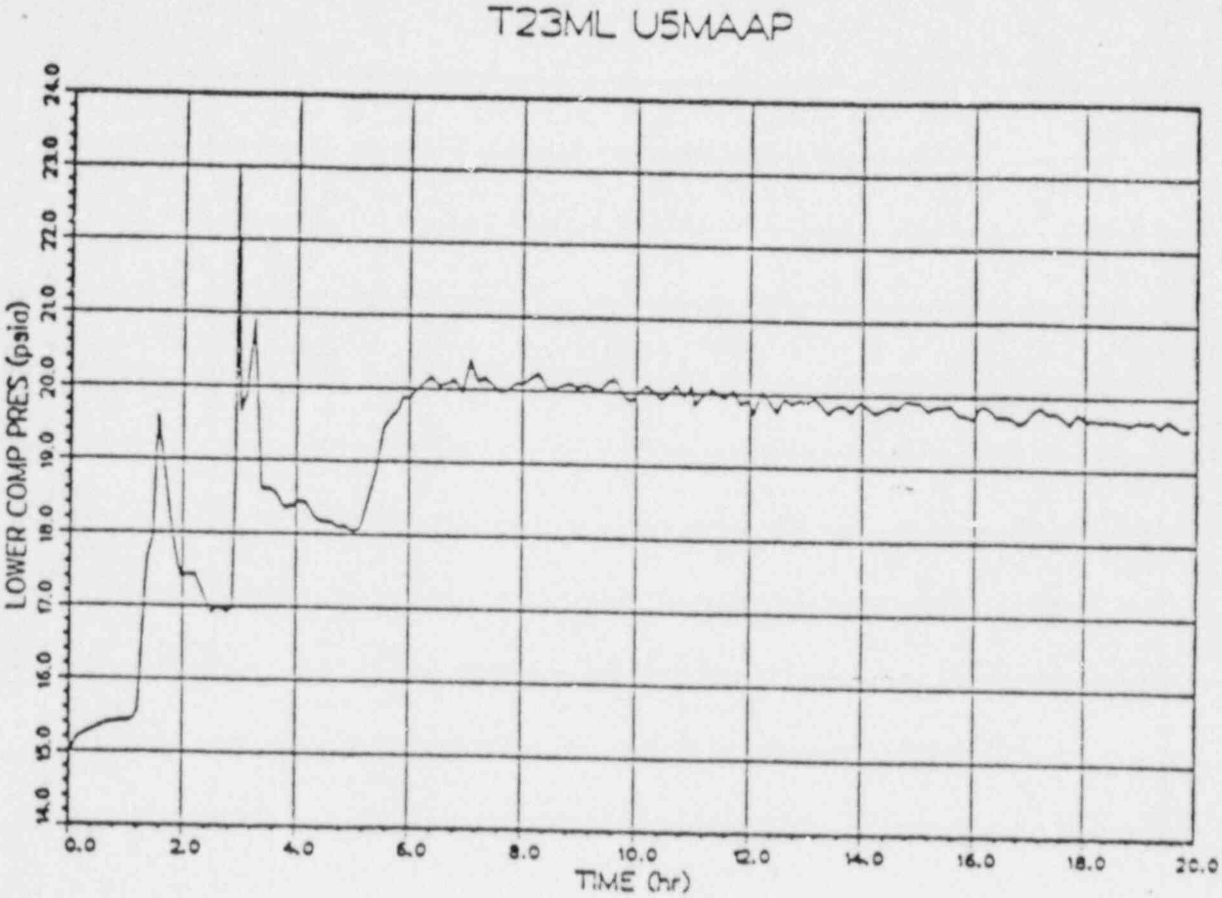


Figure A.13 Containment response for T_{23}ML accident, from IDCOR.[11]

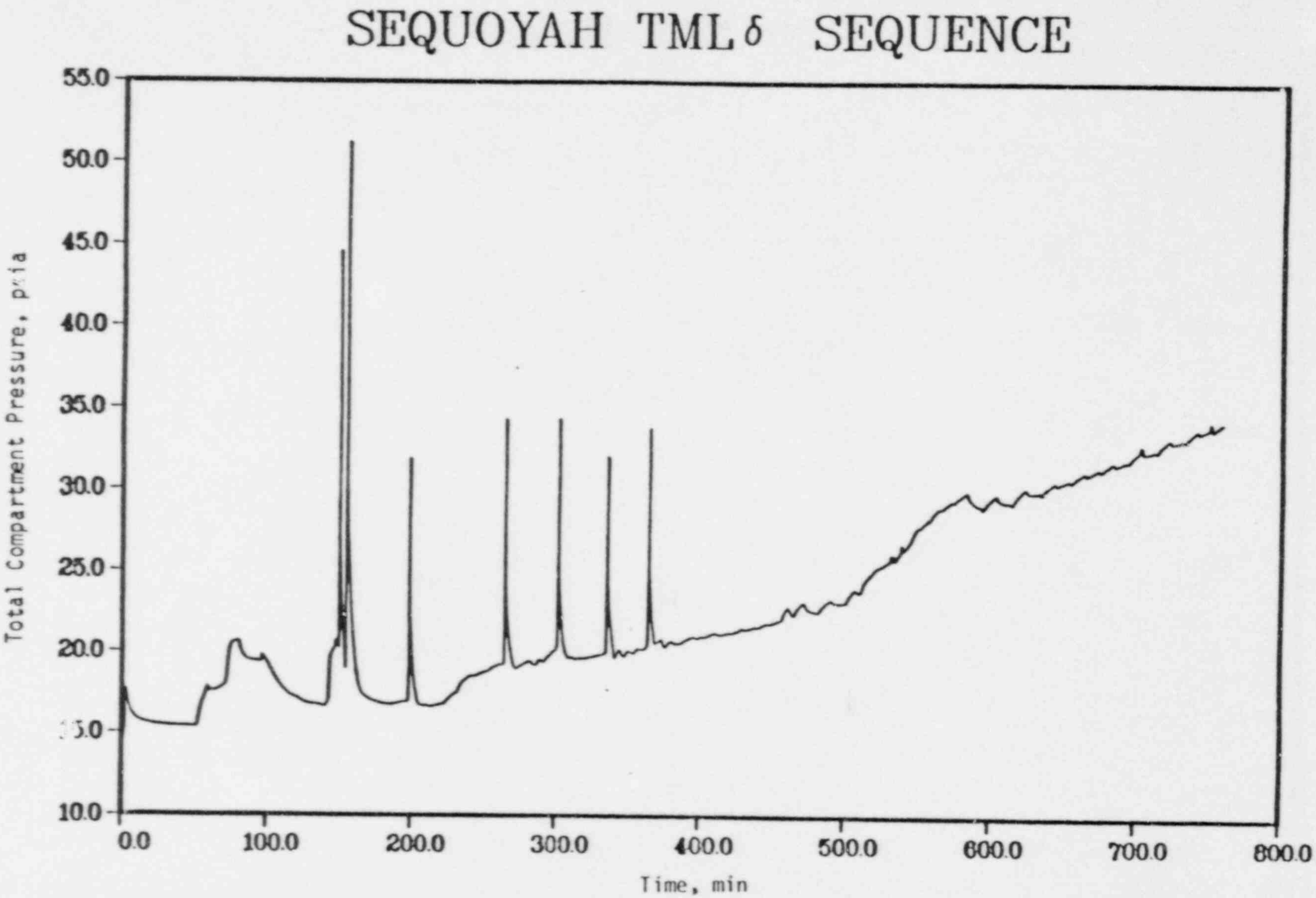


Figure A.14 Containment pressure response during Sequoyah TML-δ sequence.[10]

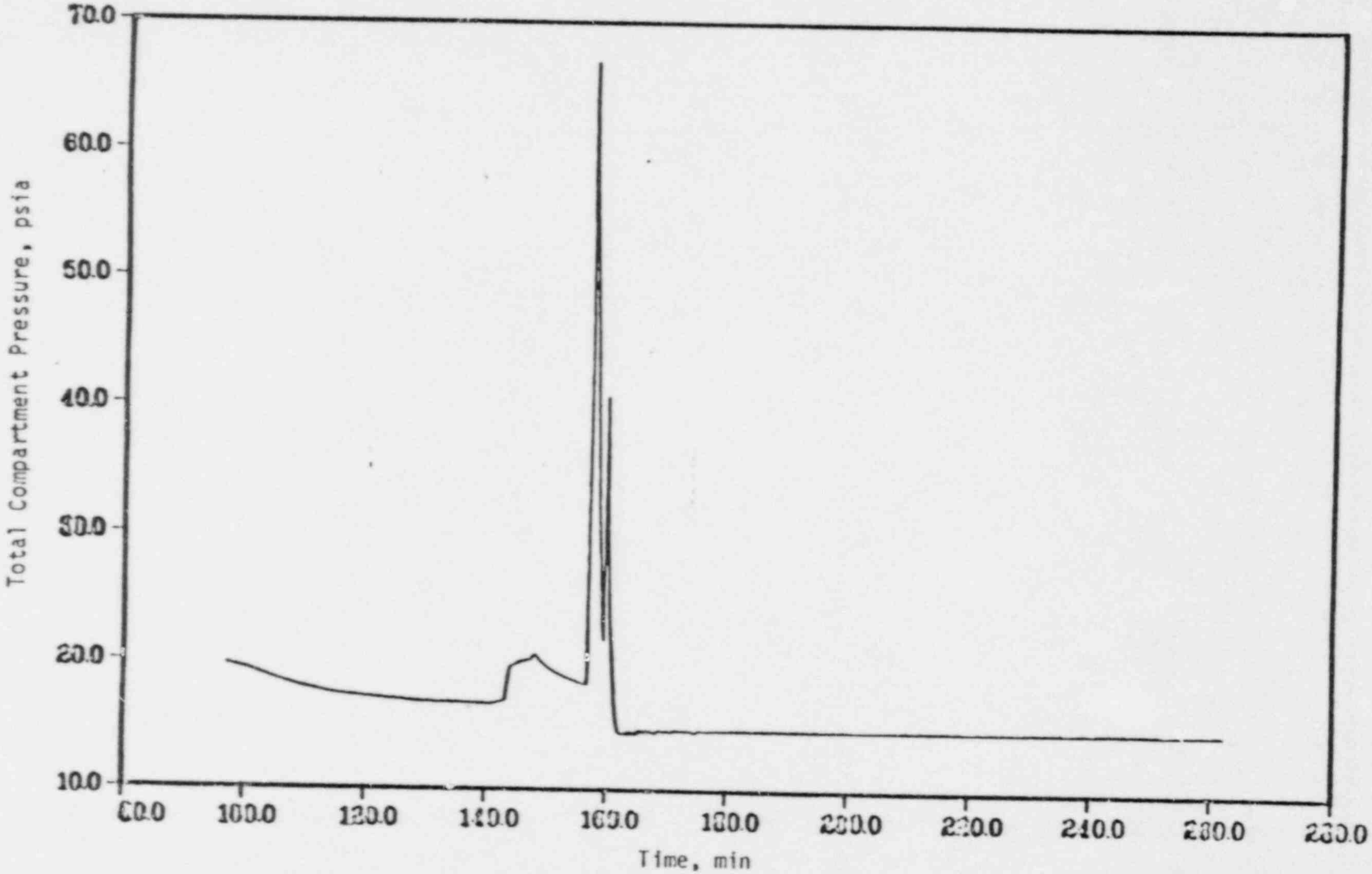


Figure A.15 Containment pressure response during Sequoyah TML-γ.[9]

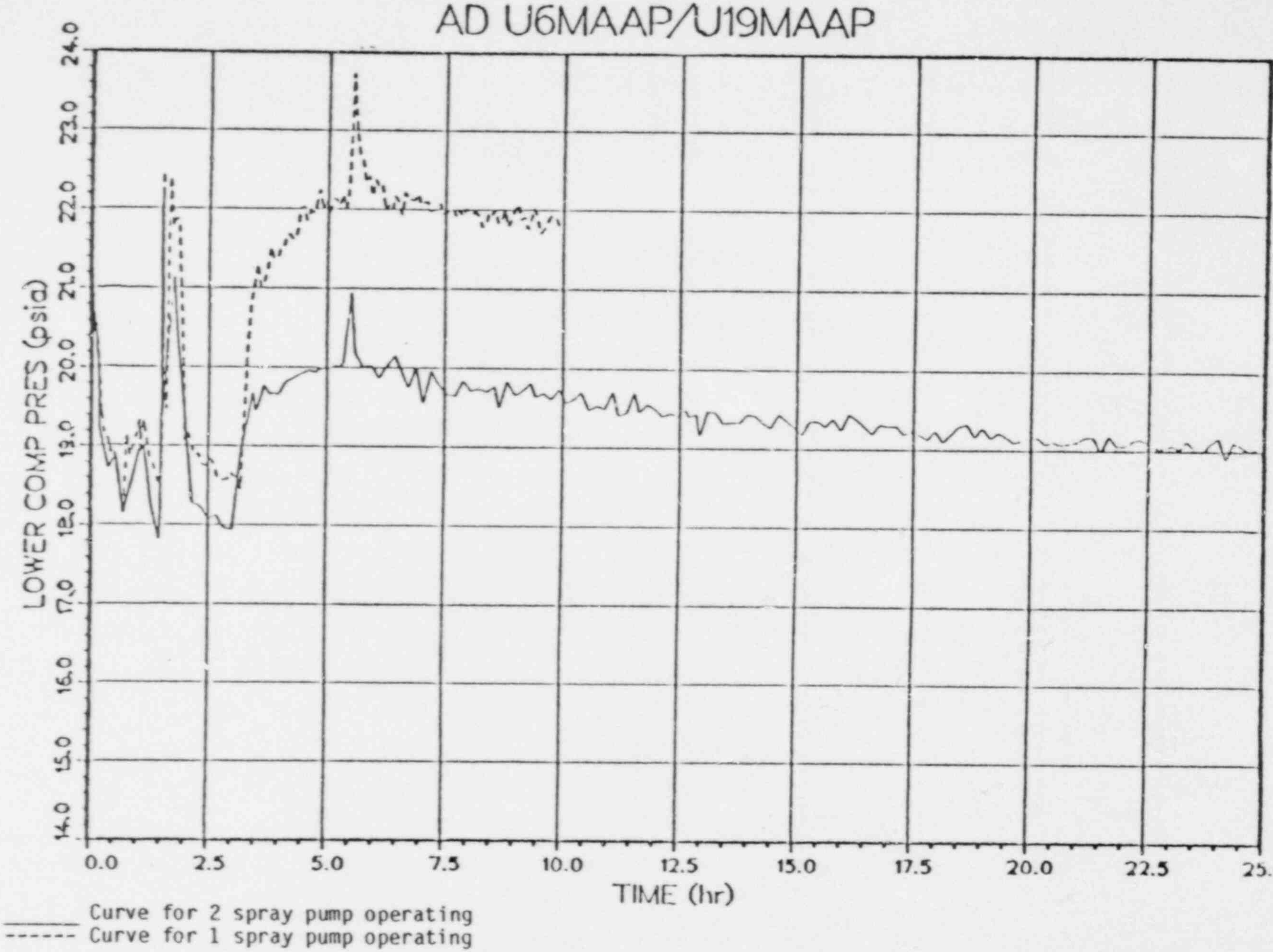


Figure A.16 Containment pressure response to AD accident.[11]

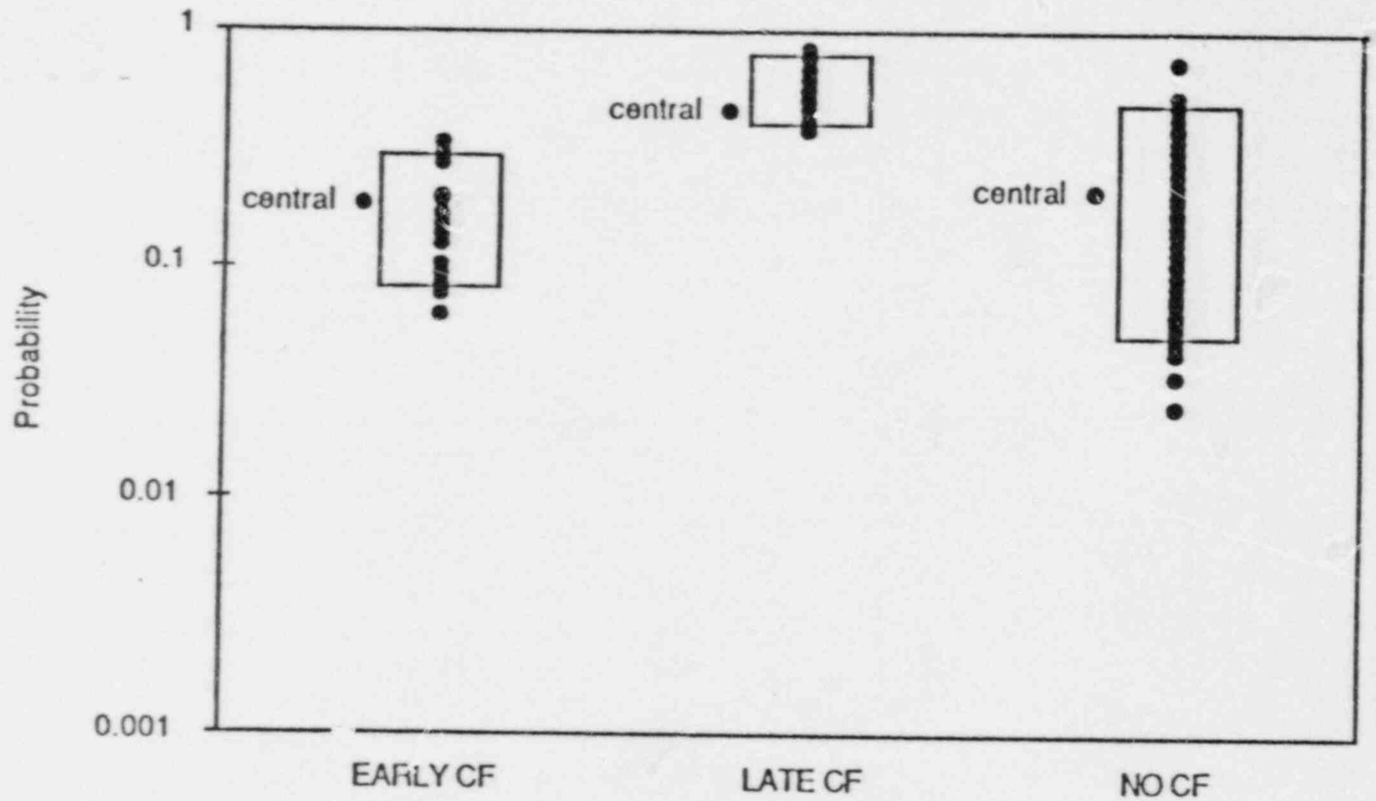


Figure A.17 Conditional probability of containment failure (SARRP) (Over all sequences).[12]

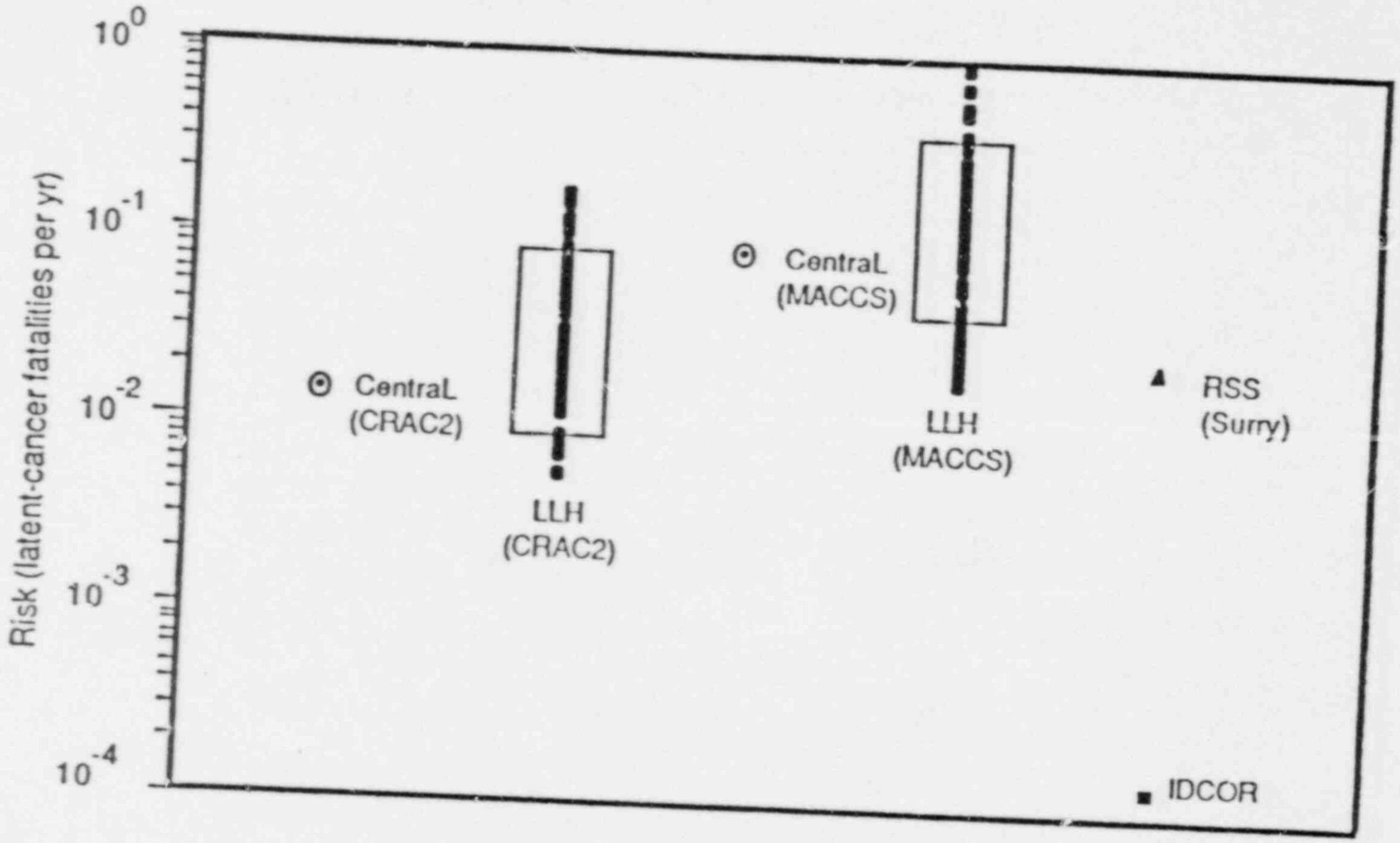


Figure A.18 Risk of latent cancer fatalities.[12]

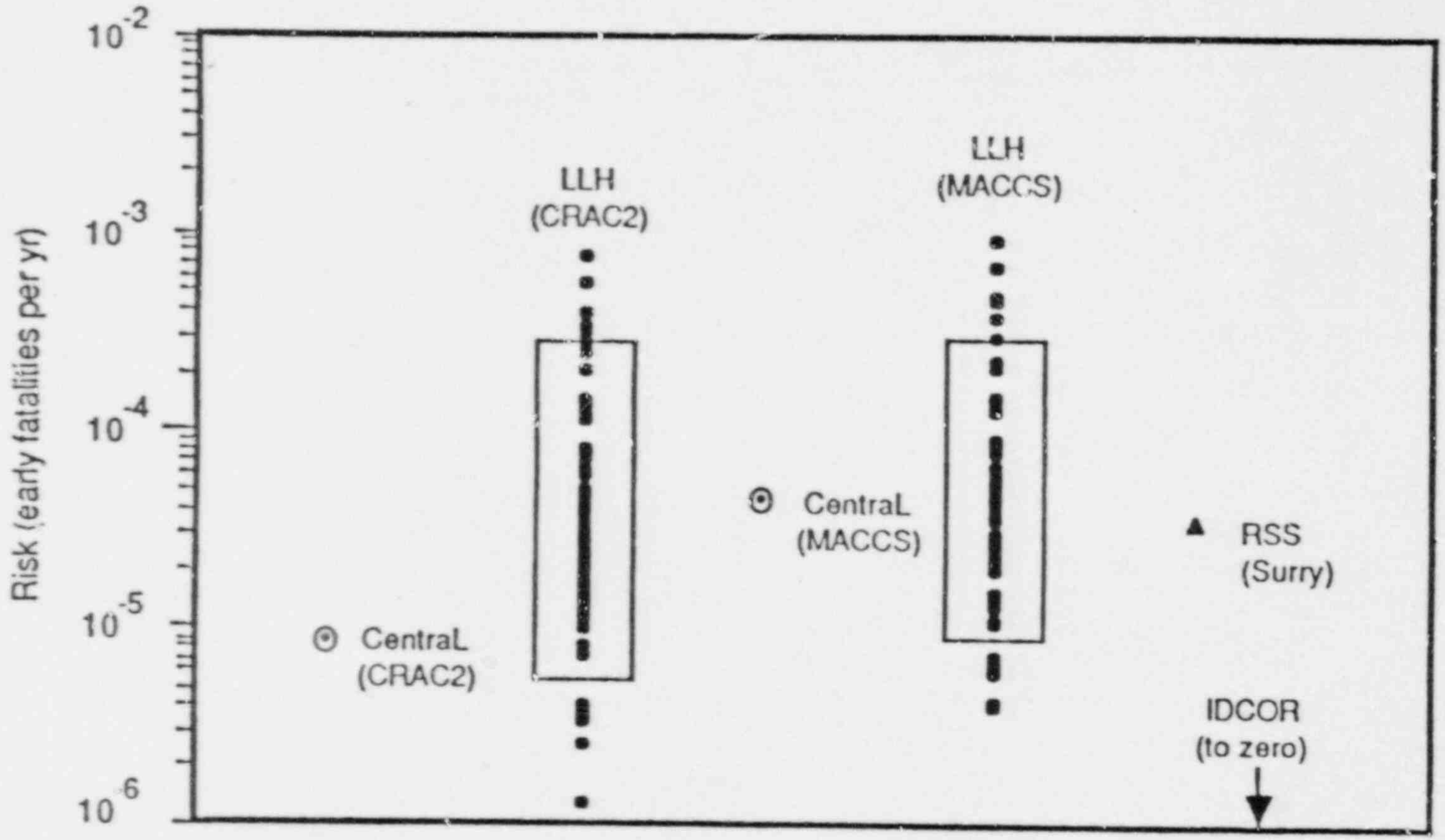


Figure A.19 Risk of early fatalities.[12]

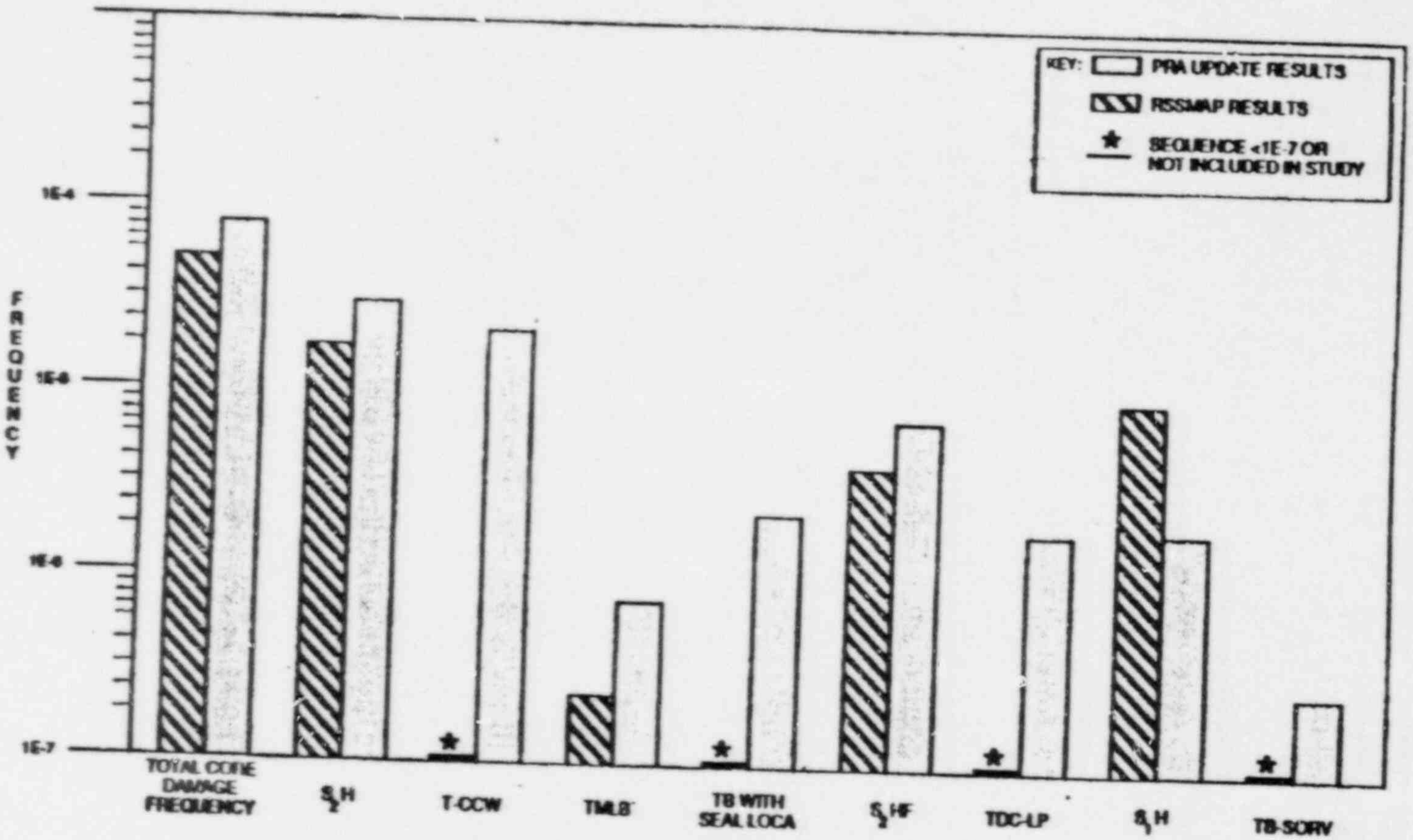


Figure A.20 Comparison of ASEP PRA update core-damage frequency with RSSMAP results for Sequoyah.[12]

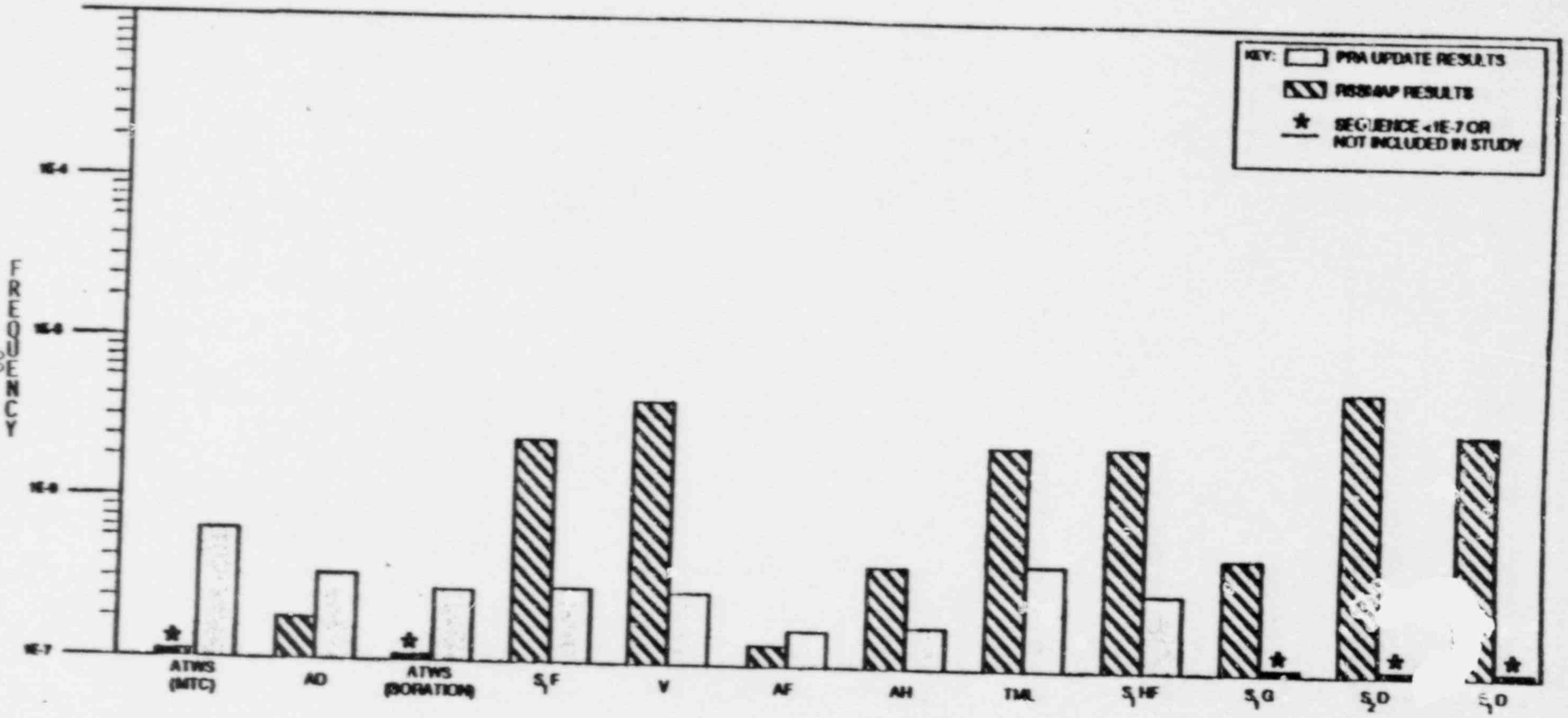


Figure A.20 (Cont'd) Comparison of ASEP PRA update core-damage frequency with RSSMAP results for Sequoyah.[12]

Table A.1 Design Comparison of Nuclear Power Plants with Ice Condenser Containments[1]

	Sequoyah	D.C. Cook	McGuire	Catawba	Watts Bar
Owner	Tennessee Valley Authority	Indiana & Michigan Electric	Duke Power	Duke Power	Tennessee Valley Authority
Site	Hamilton County, Tennessee	Berrien County, Michigan	Mecklenburg County, North Carolina	York County, South Carolina	Rhea County, Tennessee
Capacity	3411 MWt	3250 MWt	3411 MWt	3411 MWt	3411 MWt
Type	4 Loop Westinghouse PWR (2 Units)	4 Loop Westinghouse PWR (2 Units)	4 Loop Westinghouse PWR (2 Units)	4 Loop Westinghouse PWR (2 Units)	4 Loop Westinghous PWR (2 Units)
LPIS & LPRS	4 Accumulators (Cold Leg) 2 RHRS Pumps 1 Accumulator (Upper Head)	4 Accumulators (Cold Leg) 2 RHRS Pumps	4 Accumulators (Cold Leg) 2 RHRS Pumps 1 Accumulator (Upper Head)	4 Accumulators (Cold Leg) 2 RHRS Pumps 1 Accumulator (Upper Head)	4 Accumulators (Cold Leg) 2 RHRS Pumps
CSIS & CSRS	2 CS Pumps 2 ARFS Fans with Ice Condenser	2 CS Pumps 2 ARFS Fans with Ice Condenser	2 CS Pumps 2 ARFS Fans with Ice Condenser	2 CS Pumps 2 ARFS Fans with Ice Condenser	2 CS Pumps 2 ARFS Fans with Ice Condenser
HPIS & HPRS	2 SI Pumps 2 Centrifugal Charging Pumps	2 SI Pumps 2 Centrifugal Charging Pumps	2 SI Pumps 2 Centrifugal Charging Pumps	2 SI Pumps 2 Centrifugal Charging Pumps	2 SI Pumps 2 Centrifugal Charging Pumps
CHRS	2 CS HXs 2 RHR HXs	2 CS HXs 2 RHR HXs	2 CS HXs 2 RHR HXs	2 CS HXs 2 RHR HXs	2 CS HXs 2 RHR HXs
AFWS	2 Motor-Driven Pumps 1 Turbine-Driven Pump	2 Motor-Driven Pumps Shared by Both Units 1 Turbine-Driven Pump	2 Motor-Driven Pumps 1 Turbine-Driven Pump	2 Motor-Driven Pumps 1 Turbine-Driven Pump	2 Motor-Driven Pumps 1 Turbine-Driven Pump
EPS(AC)	2 Essential Power Divisions with 2 (Dedicated) + 1 (Spare) Diesel Generators	4 Essential Power Divisions with 2 (Dedicated) Diesel Generators	2 Essential Power Divisions with 2 (Dedicated) Diesel Generators	2 Essential Power Divisions with 2 (Dedicated) Diesel Generators	2 Essential Power Divisions with 2 (Dedicated) Diesel Generators
EPS(DC)	2 DC Buses (Dedicated) + 1 DC Bus (Crosstied)	2 DC Buses (Dedicated)	2 DC Buses (Dedicated)	2 DC Buses (Dedicated)	2 DC Buses (Dedicated)
CCWS	5 Pumps & 3 HXs Shared by Both Units	2 Pumps 1 Maintenance Spare Pump Shared by Both Units 2 HXs	4 Pumps 2 HXs	4 Pumps 2 HXs	5 Pumps & 3 HXs Shared by Both Units
ESWS	3 Pumps Shared by Both Units	4 Pumps Shared by Both Units	2 Pumps	2 Pumps	8 Pumps Shared by Both Units

[1]Design parameters are for a unit unless otherwise noted.

Table A.2 Initiating Event Frequencies for Sequoyah in IDCOR Study (Per Reactor Year)

Initiator Symbol	Initiator Description	Pre-IDCOR (RSSMAP)	IDCOR-Baseline & IDCOR-Committed	Source for IDCOR-Study
A	Large LOCA	4.7E-5*	1.0E-3	Zion PSS, Rev.1, Table 1.5.1-50
S_1	Intermediate LOCA	9.8E-4	1.0E-3	Zion PSS, Rev.1, Table 1.5.1-50
S_2	Small LOCA	1.8E-3	2.0E-2	Zion PSS, Rev.1 Table 1.5.1-50 & ANO-1 IREP
V	Interfacing LOCA	4.6E-6	9.0E-7	NUREG-0773
T_1	Loss-of-Offsite Power	0.2	0.027	EPRI NSAC, May (1984)
T_{23}	All Transients Other Than T_1	7.0	8.1	EPRI NP-2230

*4.7E-5 = 4.7×10^{-5}.

Table A.3 Comparisons of Dominant Core-Damage Frequencies for Pre-IDCOR, IDCOR-Baseline and IDCOR-Committed (Per Reactor Year)

Sequence	Pre-IDCOR	IDCOR-Baseline	IDCOR-Committed
S_2H	1.7E-5	7.2E-5	7.0E-6
S_1H	1.3E-5	3.7E-6	3.8E-7
S_2D	6.3E-6	5.8E-8	2.0E-8
V	4.6E-6	9.0E-7	9.0E-7
S_2HF	5.4E-6	6.0E-6	6.0E-6
S_1D	3.4E-6	3.5E-6	1.1E-7
$T_{23}ML(Z)$	3.0E-6*	2.1E-7	2.1E-7
S_1HF	2.9E-6	3.0E-7	3.0E-7
AH	2.2E-7	5.1E-7	5.1E-7
T_1B_3MLB'	3.8E-7**	8.7E-7	8.7E-7
AD	8.9E-8	1.9E-6	1.9E-6
AHF	1.4E-7	1.1E-6	1.1E-6
Total	5.7E-5	9.1E-5	1.9E-5

*Sequence $T_{23}ML$ in RSSMAP.

**Sequence $T_1B_3MLB'_{13}$ in RSSMAP.

Table A.4 Initiator Frequencies and Function Unavailabilities in Dominant Core-Damage Sequences for Sequoyah in IDCOR Study

Sequence	Event	Pre-IDCOR	IDCOR-Baseline	IDCOR-Committed
S_2H	S_2	1.8E-3	2.0E-2	2.0E-2
	H	9.6E-3	3.6E-3	3.5E-4
S_1H	S_1	9.8E-4	1.0E-3	1.0E-3
	H	1.3E-2	3.7E-3	3.8E-4
S_2D	S_2	1.8E-3	2.0E-2	2.0E-2
	D	3.5E-3	2.9E-6	1.0E-6
V	V	4.6E-6	9.0E-7	9.0E-7
S_2HF	S_2	1.8E-3	2.0E-2	2.0E-2
	HF	3.0E-3	3.0E-4	3.0E-4
S_1D	S_1	9.8E-4	1.0E-3	1.0E-3
	D	3.5E-3	3.5E-3	1.1E-4
$T_{23}ML(Z)$	T_{23}	7.0	8.1	8.1
	M	1.0E-2	1.0E-1	1.0E-1
	L	4.3E-5	4.3E-5	4.3E-5
	Z	---	6.1E-3	6.1E-3
S_1HF	S_1	9.8E-4	1.0E-3	1.0E-3
	HF	3.0E-3	3.0E-4	3.0E-4
AH	A	4.7E-5	1.0E-3	1.0E-3
	H	4.6E-3	5.1E-4	5.1E-4
T_1B_3MLB'	T_1	2.0E-1	2.7E-2	2.7E-2
	B_3	1.0E-3	1.6E-3	1.6E-3
	M	2.0E-1	1.0	1.0
	L	1.9E-2	1.9E-2	1.9E-2
	B'_{13}	5.0E-1	---	---
	B'	---	7.7E-1	7.7E-1
AD	A	4.7E-5	1.0E-3	1.0E-3
	D	1.9E-3	1.9E-3	1.9E-3
AHF	A	4.7E-5	1.0E-3	1.0E-3
	HF	3.0E-3	1.1E-3	1.1E-3

Glossary for Tables A.3 and A.4

Initiating Events

A	Large LOCA
T_1	Loss of Offsite Power Transient
T_{23}	Transients other than Loss-of-Offsite Power
S_1	Intermediate LOCA
S_2	Small LOCA
V	Interfacing Systems LOCA

Systems Failures

B_3	Diesel Generators
B'_{13} , B'	Emergency Power System Recovery
D	Emergency Coolant Injection For S_2: High-pressure injection with 1/4 charging or safety injection trains. For S_1: High-pressure injection with 1/2 charging and 1/2 safety injection trains. For A: Low-pressure safety injection with 1/2 trains and 2/4 accumulators.
F	Containment Spray Recirculation
H	Emergency Coolant Recirculation For S_1 & S_2: 1/2 low-pressure trains and high-pressure recirculation with 1/2 charging or 1/2 safety injection trains. For A: Low-pressure recirculation with 1/2 trains and hot leg recirculation.
L	Emergency Feedwater System, auxiliary feedwater to 2/4 steam generators.
M	Main Feedwater System (Normal Operation)
(Z)	Feed and Bleed Cooling

Table A.5 Initiating Event Frequencies for Sequoyah in ASEP Study (Mean Values Per Reactor Year)

Initiator Symbol	Initiator Description	Annual Frequency	Source
A	Large LOCA	5.0E-4	Past PRAs
S_1	Intermediate LOCA	1.0E-3	Past PRAs
S_2	Small LOCA	2.0E-2	Past PRAs
V	Interfacing LOCA	3.3E-7	Ref. 3
T_1	Loss of Offsite Power	7.0E-2	NUREG-1032
T_2	Transients with PCS Initially Unavailable	3.0	NUREG-3862
T_3	Transients with PCS Initially Available	4.8	NUREG-3862
T_{DCX}	Loss of dc Bus "X"	9.0E-4	Ref. 3
T_{CCW}	Loss of CCWS	2.7E-5	Ref. 3

Table A.6 Dominant Core-Damage and Core Vulnerable Frequencies for Sequoyah in ASEP Study (Point Mean Values Per Reactor Year)

Sequence	Core Damage[a]	Core Vulnerable[a]	Core Damage[b]
AH_1	2.7E-6		2.0E-7
AH_1F	1.9E-7		1.9E-7
AF		2.2E-6	1.9E-7
AD_5	3.8E-7		3.9E-7
AD_6	1.5E-7		1.6E-7
AD_6F	1.6E-8		ε[c]
AG_1			4.3E-8
S_1H_4	5.8E-6		7.9E-7
S_1H_4F	2.6E-7		3.8E-7
S_1H_2	2.1E-6		1.6E-6
S_1H_2F	2.6E-7		ε
S_1F		4.4E-6	3.8E-7
S_1D_2	5.4E-7		ε
$S_1D_2H_4F$	1.5E-7		ε
S_1G_1			8.5E-8
S_2H_2F	3.5E-7 (6.3E-8)[d]		3.1E-7
S_2H_2	4.2E-5 (6.2E-6)[d]		2.9E-5
$S_2H_3G_1$	1.1E-7 (ε)[d]		ε
S_2H_3	2.6E-5		9.7E-6
S_2H_3F	6.4E-6		7.7E-6
S_2D_1	4.0E-7		ε
$S_2D_1H_3F$	2.3E-6		ε
S_2G_2F		1.1E-7	ε
V			3.3E-7

Table A.6 (Continued)

Sequence	Core Damage[a]	Core Vulnerable[a]	Core Damage[b]
$T_1L_1P_1$	9.4E-7		ε
$T_1L_1H_3$	2.6E-6		ε
$T_1L_1H_2$	8.4E-9		ε
$T_1L_1D_1F$			8.3E-7
$T_1L_1D_1H_3F$	1.5E-4		ε
$T_1L_1D_1H_3$	2.0E-9		ε
$T_1D_3WD_1F$			2.9E-6
$T_1Q_1\text{-}D_1F$			3.3E-7
$T_2L_1MP_1$	5.7E-7		5.2E-7
$T_2L_1MH_3$	2.9E-8		ε
$T_2L_1MH_2$	3.0E-8		ε
$T_3L_1MP_1$	7.5E-8		ε
$T_3L_1MH_3$	1.7E-9		ε
$T_3L_1MH_2$	8.1E-9		ε
$T_{DCI}L_1D_1$	1.4E-9		ε
$T_{DCII}L_1D_1$	1.2E-7		ε
$T_{DCI}L_1P_1$	2.2E-6		1.1E-6
$T_{DCII}L_1P_1$	2.2E-6		1.1E-6
$T_{DCI}D_3WD_1H_3$	2.3E-7		ε
$T_{DCII}D_3WD_1H_3$	1.0E-5		ε
T_{CCW}			2.7E-5
$TK_1K_2X_1$			6.8E-7
$TK_1K_2D_4$			4.1E-7
Total	2.6E-4 (2.2E-4)[d]	6.7E-6	8.6E-5 (1.0E-4)[e]

[a]From information presented to QC meeting, December 1985 (Ref. 2).

[b]From information contained in the draft ASEP report, December 1986 (Ref. 3).

[c]Not provided as dominant sequences (<1.0E-7).

[d]Results obtained from applying β-factors and recoveries to cutsets for S_2H_2F, S_2H_2, and $S_2H_3G_1$ (Ref. 2).

[e]Mean core-damage frequency obtained from Monte Carlo simulation.

Glossary for Table A.6

A	Large LOCA (>6" diameter).
D_1	High-pressure injection with 1/4 charging or safety injection trains (S_2 or transients only).
D_2	High-pressure injection with 2/4 charging or safety injection trains (S_1 only).
D_3	Seal injection flow to reactor coolant pumps from 1/2 charging trains.
D_4	Emergency boration from 1/2 charging trains (ATWS only).
D_5	Cold leg accumulators with 3/3 intact trains.
D_6	Low-pressure safety injection with 1/2 trains.
F	Containment spray recirculation with 1/2 trains.
G_1	Containment heat removal with 1/2 containment spray recirculation trains.
G_2	Containment heat removal with 1/2 low pressure recirculation trains.
H_1	Low-pressure recirculation with 1/2 trains and hot leg recirculation.
H_2	High-pressure recirculation with 1/2 charging or safety injection trains.
H_3	Low-pressure injection in miniflow mode (1/2 trains) and low pressure recirculation (1/2 trains) (hot leg not required).
H_4	Low-pressure injection miniflow mode (1/2 trains) and low pressure recirculation with hot leg recirculation (1/2 trains).
K_1	Automatic reactor trip.
K_2	Manual reactor trip.
L_1	Auxiliary feedwater to 2/4 steam generators.
L_2	Auxiliary feedwater to 3/4 steam generators (ATWS with MTC between -20 and -7 pcm only) or 2/4 steam generators (ATWS with low MTC < -20 pcm).
M	Main feedwater or condensate feedwater trains.
N	Turbine trip (ATWS only).
P_1	PORVs and block valves open (2/2 trains) for feed and bleed.
P_2	Primary safety valves open (3/3) (ATWS only).

Glossary for Table A.6 (Cont'd)

Q_1	All relief valves reclose.
Q_2	All safety and relief valves reclose (ATWS only).
S_1	Intermediate LOCA (2" to 6" diameter).
S_2	Small LOCA (3/8" to 2" diameter).
T_1	Loss-of-offsite power.
T_2	Transient with initial loss of power conversion system and main feedwater.
T_3	Transient with power conversion system initially available.
T_{DCX}	Loss of 125 V dc bus "X".
W	Component cooling water to reactor coolant pump thermal barriers.
X_1	Unfavorable moderator temperature coefficient (< -7 pcm).
X_2	Low moderator temperature coefficient (< -20 pcm).
Z	Ice-condenser system.

Table A.7 Comparison of Initiating Event Frequencies Used in IDCOR and ASEP Studies (Mean Values Per Reactor Year)

Initiator	Initiator Frequency	
	IDCOR[a]	ASEP
A	1.0E-3	5.0E-4
S_1	1.0E-3	1.0E-3
S_2	2.0E-2	2.0E-2
V	9.0E-7	3.3E-7
T_1	2.7E-2	7.0E-2
T_2	8.1[b]	3.0
T_3	b	4.8
T_{DCX}	c	9.0E-4
T_{CCW}	c	2.7E-5

[a]For both baseline and committed.
[b]8.1 events include all transients other than loss of offsite power (T_1).
[c]IDCOR does not specifically consider these initiating events.

Table A.8 Comparison of Dominant Core-Damage Frequencies Assessed in IDCOR and ASEP Studies (Per Reactor Year)

Initiator	Sequoyah CDF		
	IDCOR[a] Baseline	IDCOR[a] Committed	ASEP[b]
A	3.5E-6	3.5E-6	1.2E-6
S_1	7.5E-6	7.9E-7	3.2E-6
S_2	7.8E-5	1.3E-5	4.7E-5
V	9.0E-7	9.0E-7	3.3E-7
T_1	8.7E-7	8.7E-7	4.1E-6
T_2	2.1E-7[c]	2.1E-7[c]	5.2E-7
T_3			ε
T_{DCX}			2.2E-6
T_{CCW}			2.7E-5
(ATWS)	d	d	1.1E-6
Total Core-Damage Frequency	9.1E-5	1.9E-5	8.6E-5 (1.0E-4)[e]

[a]"Mixed"-point estimates. Common cause failures and operator recoveries were included.

[b]From information contained in the draft ASEP report, December 1986 (Ref. 3); mean-point estimates.

[c]From all transients other than loss of offsite power (T_1).

[d]Not presented as dominant sequences.

[e]Mean core-damage frequency obtained from Monte Carlo simulation.

Table A.9. Review of Accident Analyses for Sequoyah Nuclear Plant

IDCOR		SARP	
Sequence Analyzed	Containment Failure Time (min)	Sequence Analyzed	Containment Failure Time (min)
S_2D-δ	NCF[11]	S_2D-δ	703[4]
S_2H-δ	NCF[11]	S_2H-δθ	798[4]
		S_2H-γθ	110[4]
		S_2H-γθ	NCF[10]
S_2HF-δ Drains blocked	1556[11]	S_2HF-δ	NCF[10]
		S_3HF_2-γ, S_3HF_3-γ	412[10]
		S_2HF-γ	352[9]
		S_2HF-δ	301[4]
		S_2HF-α	180[4]
S_2HF-δ Drains open S_1 not analyzed	572[11]	S_2HF-δθ	197[4]
		S_3HF_1-γ	412[10]
		S_1HF-δ	219[4]
TMLB'-δ	1628[11]	TMLB'-δ	553[9]
		TMLB'-γ	158[9]
		TMLB'-δ	328[4]
		TMLB-δθ	244[4]
		TM-S_3B	374[10]
T_{23}ML-δ	NCF[11]	TML-δ	NCF[10]
		TML-γ	157[9]
		TML-γ	238[4]
		TML-δ	660[4]
		TMLU-STGR	127[10]
AD-δ	NCF[11]	AD-δ	361[4]
		AD-γ	66[4]
		AD-α	20[4]
		AHF-δ	219[4]

Notes: NCF = no containment failure.
RSS notation (see Table 2.21).

Table A.10 Containment Response Comparison from IDCOR and RSSMAP Calculations for S_2D and S_2H Accidents

	Sequences Analyzed				
	S_2D		S_2H		
	IDCOR[11]	RSSMAP[4]	IDCOR[11]	RSSMAP[4]	
Event Time (Hours)	δ	δ	$\delta\theta^a$	$\delta\theta^a$	$\gamma\theta^a$
Core Uncovery	0.8	0.77	1.3	1.03	1.03
Start of Core Damage	1.0	NA	1.4	NA	NA
Start of Core Melt	1.4	1.03	1.9	1.33	1.33
Fraction of Clad Reacted	0.32	NA	0.33	NA	NA
Time of RPV Failure	2.81	2.68	3.31	1.83	1.83
Time of Ice Depletion	4.92	2.82	4.55	2.0	1.97
Peak Containment Pressure (psia)	21.5	30.0	21.4	30.0	30.0
Time of Peak Pressure	2.81	11.72	3.31	13.3	1.83
Containment Failure Time	NCF	11.72	NCF	13.3	1.83
Containment Failure Cause	None	Pressure	None	Pressure	Hydrogen Burn
Start of Concrete Melt	NA	3.03	NA	4.33	4.33

[a]Water in the cavity at RPV failure time.
NA = Not available.
NCF = No containment failure.

Table A.11 Containment Response Comparison from IDCOR, RSSMAP and SARP Calculations for S_2HF Accidents

	Sequences Analyzed					
	S_2HF (Drains Open)		S_3HF_1 (Drains Open)		S_2HF (Drains Blocked)	
Event Time (Hours)	IDCOR[11] $\delta\theta^a$	RSSMAP[4] $\delta\theta^a$	SARP[10] γ	IDCOR* δ	RSSMAP[4] δ	SARP[9] γ
Core Uncovery	1.2	2.13	4.54	1.2	2.13	2.72
Start of Core Damage	1.4	NA	NA	1.4	NA	NA
Start of Core Melt	1.9	2.53	6.06	1.9	2.53	3.27
Fraction of Clad Reacted	0.35	NA	0.74	0.34	NA	0.66
Time of RPV Failure	3.31	3.22	6.84	3.35	3.22	4.33
Time of Ice Depletion	4.36	3.25	16.53	3.81	4.55	3.30
Peak Containment Pressure (psia)	65.0	30.0	62.0	65.0	30.0	121
Time of Peak Pressure	9.54	3.28	6.87	25.9	5.02	5.87
Containment Failure Time	9.54	3.28	6.87	25.9	5.02	5.87
Containment Failure Cause	Pressure	Pressure	Hydrogen Burn	Pressure	Pressure	Hydrogen Burn
Start of Concrete Melt	NA	5.23	6.84	NA	3.22	5.87
Time of 30.0 psia Pressure	5.2	3.28	3.35	4.0	5.02	3.3

[a]Water in the cavity at RPV failure time.
NA = Not available.
*Table 7.1-1 from Ref. 11 and Table 7-5 from Ref. 1.

Table A.12 Containment Response Comparison from IDCOR, RSSMAP and SARP Calculations for TMLB', $T_{23}ML$ Accidents

Event Time (Hours)	TMLB'				$T_{23}ML$				AD	
	IDCOR*	SARP[9]	SARP[9]	RSSMAP[4]	IDCOR*	SARP[9]	SARP[9]	RSSMAP[4]	IDCOR*	RSSMAP[4]
	δ	δ	γ	δ			γ	δ	δ	δ
Core Uncovery	1.8	1.63	1.63	3.07	1.7	1.62	1.62	3.07	0.4	0.02
Start of Core Damage	2.0	NA	NA	NA	1.9	NA	NA	NA	0.8	NA
Start of Core Melt	2.5	2.03	2.03	3.33	2.4	2.02	2.02	3.33	1.1	0.1
Fraction of Clad Reacted	0.30	0.49	0.49	NA	0.26	0.49	0.49	NA	0.42	NA
Time of RPV Failure	3.35	2.63	2.63	3.97	2.91	2.59	2.62	3.35	1.52	1.1
Time of Ice Depletion	5.84	9.27	7.62	5.80	5.36	8.85	NA	>16.7	3.15	>17.0
Peak Containment Pressure (psia)	65.0	60.0	85.0	30.0	23.0	52.0	67.5	30.0	22.0	30.0
Time of Peak Pressure	27.1	9.21	2.63	5.47	2.9	2.59	2.62	11.0	5.3	6.0
Containment Failure Time	27.1	9.21	2.63	5.47	NCF	NCF	2.62	11.0	NCF	6.0
Containment Failure Cause	Pressure	Pressure	Hydrogen Burn	Pressure	None	None	Hydrogen Burn	Pressure	None	Pressure
Start of Concrete Melt	3.35	2.69	2.68	3.77	NA	2.59	2.62	3.35	NA	4.48
Time of 30.0 psia Pressure	3.5	3.16	2.62	5.47	NA	2.50	2.61	11.0	NA	6.0

NA = Not available.

NCF = No containment failure.

*Table 7.1-1 from Ref. 11 and Table 7-5 from Ref. 1.

Table A.13 Containment Response Comparison from IDCOR and SARP Calculations for V, TMLU-SGTR, β/W

Event Time (Hours)	Sequences Analyzed: V		TMLU-SGTR	β/W	$S_2H-\beta$
	IDCOR*	RSSMAP[9]	SARP[10]	IDCOR*	IDCOR*
Core Uncovery	NA	0.38	1.73	NA	NA
Start of Core Damage	NA	NA	NA	NA	NA
Start of Core Melt	NA	0.63	2.12	NA	NA
Fraction of Clad Reacted	NA	NA	0.43	NA	NA
Time of RPV Failure	20.0	1.52	2.82	7.0	7.0
Time of Ice Depletion	NA	NA	NA	NA	NA
Peak Containment Pressure (psia)	NA	NA	22.4	NA	NA
Time of Peak Pressure	NA	NA	2.82	NA	NA
Containment Failure Time	0.0	0.0	2.57	0.0	0.0
Containment Failure Cause	Isolation Failure	Isolation Failure	SG Tubes Ruptured (at core slump)	Isolation Failure	Isolation Failure
Start of Concrete Melt	NA	1.52	2.82	NA	NA

*Table 7-6 of Ref. 1.

Table A.14 Sequoyah Release Fraction in Environment - IDCOR-SARP Comparison of δ Cases for S_2HF, TMLB and TML Accidents

Accident Analyzed	Containment Failure (Hours)	Fission Product Category (Fraction of Core Inventory)					References
		CsI	Te	Sr	Ru	CsOH	
S_2HF-δ Drains Open	9.54	$<10^{-5}$	1.7×10^{-5}	$<10^{-5}$	$<10^{-5}$	2.9×10^{-5}	IDCOR[11]
S_2H-δ Drains Open	10	7.4×10^{-4}	1.5×10^{-4}	NA	NA	7.9×10^{-4}	IDCOR[1]
S_2HF-δθ Drains Open	3.3	4.5×10^{-2}	1.3×10^{-1}	3.3×10^{-2}	1.5×10^{-2}	2.6×10^{-1}	RSSMAP[4]
S_2HF-δ Drains Closed	25.9	2.1×10^{-5}	$<10^{-5}$	$<10^{-5}$	$<10^{-5}$	6.9×10^{-5}	IDCOR[11]
S_2HF-δ Drains Closed	24	6.5×10^{-4}	1.6×10^{-4}	NA	NA	6.5×10^{-4}	IDCOR[1]
S_2HF-δ Drains Closed	5.0	1.0×10^{-2}	3.9×10^{-1}	2.4×10^{-2}	2.4×10^{-2}	2.3×10^{-1}	RSSMAP[4]
TMLB'-δ	27.1	5.0×10^{-4}	2.6×10^{-5}	$<10^{-5}$	$<10^{-5}$	6.4×10^{-4}	IDCOR[11]
TMLB'-δ	27	4.1×10^{-3}	2.5×10^{-3}	NA	NA	4.1×10^{-3}	IDCOR[1]
TMLB'-δ	9.25	3.9×10^{-4}	2.0×10^{-3}	NA	NA	4.5×10^{-4}	SARP[9]
$T_{23}ML$-δ	NCF	0.0	0.0	0.0	0.0	0.0	IDCOR[11] 1
TML-δ	NCF	6.9×10^{-9}	1.6×10^{-8}	NA	NA	7.4×10^{-9}	SARP[9]
AD-δ	NCF	0	0	0	0	0	IDCOR[11]
AD-δ	6.0	7.0×10^{-7}	3.0×10^{-7}	3.0×10^{-8}	2.0×10^{-8}	3.0×10^{-7}	RSSMAP[4]

Table A.15 Sequoyah Release Fraction in Environment - Comparison of γ-Cases With β, V cases

Accident Analyzed	Containment Failure (Hours)	Fission Product Category (Fraction of Core Inventory) CsI	Te	Sr	Ru	CsOH	References
$S_3HF_1-\gamma$ Drains Open	6.87	1.6×10^{-2}	4.4×10^{-3}	2.6×10^{-4}	2.0×10^{-8}	1.3×10^{-2}	SARP [10]
$S_2HF-\gamma$ Drains Closed	5.87	3.3×10^{-2}	5.5×10^{-2}	4.9×10^{-2}	4.2×10^{-4}	3.2×10^{-2}	SARP [9]
$S_3HF_3-\gamma$ Drains Closed	6.87	1.4×10^{-2}	5.0×10^{-3}	1.0×10^{-3}	1.4×10^{-7}	1.1×10^{-2}	SARP [10]
$S_2HF-\gamma$	3.28	1.3×10^{-1}	4.9×10^{-1}	6.8×10^{-2}	4.2×10^{-2}	5.7×10^{-1}	RSSMAP [4]
TMLB'-γ	2.63	1.7×10^{-2}	1.4×10^{-2}	NA	NA	2.3×10^{-2}	SARP [9]
TML-γ	2.62	1.3×10^{-3}	5.5×10^{-4}	NA	NA	7.0×10^{-3}	SARP [9]
TML*-γ	4.0	1.6×10^{-2}	3.1×10^{-1}	5.0×10^{-3}	1.9×10^{-2}	8.0×10^{-2}	RSSMAP [4]
V Case	0.0	5.3×10^{-1}	3.2×10^{-1}	5.8×10^{-2}	3.0×10^{-2}	5.0×10^{-1}	RSSMAP [4]
V Case	0.0	1.0×10^{-4}	1.0×10^{-4}	NA	NA	1.0×10^{-4}	IDCOR [1]
β/W Case	0.0	1.6×10^{-2}	4.0×10^{-3}	NA	NA	1.6×10^{-2}	IDCOR [1]
$S_2HF-\beta$	0.0	2.1×10^{-3}	2.4×10^{-3}	NA	NA	2.1×10^{-3}	IDCOR [1]

*Decontamination factor 100 in ice bed assumed.

Table A.16 MARCH Results[4] for Ice Condenser PWR Accident Sequences[a]

SEQUENCE	ECCS		CSS		CORE	CORE MELT		VESSEL	ICE MELT	CONT	CONCRETE
	START	STOP	START	STOP	UNCOVERY	START	END	FAILURE	COMPLETE	FAILURE	MELT START
AD-α	-	-	1	-	1	5	20	46	b	20	46
AD-γ	-	-	1	66	1	5	18	66	>1000	66	67
AD-δ	-	-	1	361	1	5	18	66	>1000	361	67
AHF-δ	1	129	1	129	177	202	225	269	121	219	269
S_1HF-δ	1	126	1	126	155	177	215	259	159	219	269
S_2D-δ	-	-	1	703	46	62	118	161	169	703	182
S_2H-γθ	1	50	1	110	62	80	96	110	118	110	260
S_2H-δθ	1	50	1	798	62	80	96	110	120	798	260
S_2HF-α	1	109	1	109	128	152	180	206	b	180	324
S_2HF-δ	1	109	1	109	128	152	180	193	273	301	193
S_2HF-δθ	1	109	1	109	128	152	180	193	195	197	314
TMLB'-δ	-	-	-	-	184	200	232	238	348	328	238
TMLB'-δθ	-	-	-	-	184	200	232	238	242	244	384
TML-γ	238	-	1	238	184	200	232	238	>1000	238	238
TML-δ	238	-	1	660	184	200	232	238	>1000	660	238
V	-	-	-	-	23	38	57	91	-	-	91

[a]All times in minutes

[b]Ice bed bypassed after steam explosion

Table A.17 Summary of Ice Condenser PWR CORRAL Results

Sequence	Cumulative Fractions of Core Inventory Released to the Atmosphere[a]						
	Xe-Kr	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La
AD-α[b c]	1.0	0.23	0.37	0.42	0.045	0.41	0.003
AD-γ[b]	1.0	5(4)	6(4)	0.003	4(5)	2(4)	3(5)
AD-γ[d]	1.0	0.002	0.006	0.026	3(4)	0.002	3(4)
AD-δ[b]	1.0	7(7)	3(7)	3(7)	3(8)	2(8)	4(9)
S_2H-γθ[b]	1.0	0.005	0.040	0.15	0.003	0.009	0.002
S_2HF-α	1.0	0.27	0.68	0.41	0.081	0.43	0.003
S_2HF-γ	1.0	0.13	0.57	0.49	0.068	0.042	0.007
S_2HF-δ	1.0	0.010	0.23	0.34	0.024	0.024	0.004
S_2HF-δθ	1.0	0.045	0.26	0.13	0.033	0.015	0.002
TMLB'-δ[b]	1.0	0.007	0.014	0.050	0.001	0.003	6(4)
TMLB'-δθ[b]	1.0	0.063	0.11	0.089	0.013	0.008	0.001
TMLB'-δθ[d]	1.0	0.063	0.14	0.095	0.017	0.009	0.001
TML-γ[b]	1.0	0.016	0.080	0.31	0.005	0.019	0.004
V[e]	1.0	0.77	0.80	0.51	0.093	0.049	0.007
V[f]	1.0	0.53	0.50	0.32	0.058	0.030	0.004
V[f g]	1.0	0.48	0.42	0.086	0.052	0.016	0.002

[a]The notation 5(5) is an abbreviation for 5×10^{-5}.

[b]Ice bed decontamination factor of 100 for iodine and particulates until 90% of the ice has melted.

[c]Ice bed bypassed after the steam explosion.

[d]Ice bed decontamination factor of 10 for iodine and particulates until 90% of the ice has melted.

[e]Assuming release through the UHI equipment room.

[f]Assuming release through the auxiliary building.

[g]With the air return fans and ice condenser operating after reactor vessel melt-through.

Table A.18 SARP-IDCOR Comparison of Input Data

	SARP[9]	IDCOR[11]	IDCOR/SARP % Data or Comment
Reactor Power, MWt	3,570	3,409	95.5%
Operating Pressure, psia	2,265	2,250	SM
Operating Temperature, F	580	578.2	SM
Primary System Volume, ft^3	13,195	Proprietary	U
Primary System Water Inventory, lb.	547,400	Proprietary	U
Active Core Height, ft.	12	12	SM
Core Flow Area, ft^2	51.5	56	SM
Total Vessel Water Area, ft^2	107.0	Proprietary	U
Pressurizer Relief Valve Setpoint, psia	2,350	2,350	SM
Pressurizer Relief Valve Capacity, lb/min	21,000	3,393 x 4 loops x 2 valves	129%
Steam Generator Water Inventory, lb	350,000	94,877 x 4 loops	108.4%
Steam Generator Relief Valve Setpoint, psig	1,100	1,105 average	SM
Zircaloy in Core, lb.	50,913	46,993	SM
Misc Metal in Core, lb.	19,158	Proprietary	U
UO_2 in Core, lb.	222,739	222,645	SM
Weight of Grids Included in Debris, lb.	95,000	Proprietary	U
Bottom Head Diameter, ft.	14.4	13,916	SM
Bottom Head Thickness, ft.	0.469	0.479	SM
Fuel Rods in Core	50,952	50,952	SM
Rod Diameter, in.	0.374	0.374	SM
Fuel Diameter, in.	0.3226	0.3624	112%

Table A.18 (Continued)

	SARP[9]	IDCOR[11]	IDCOR/SARP % Data or Comment
Containment Parameters			
Total Free Volume, ft^3	1,285,580	1,202,700 (with empty ice condenser)	SM
Upper Compartment, ft^3	897,880	808,800 (U.C. + ice condenser)	SM
Lower Compartment, ft^3	387,700	394,200 (Comp. D + cavity)	SM
Initial Temperature Upper Compartment, F	100	85	15F lower
Initial Temperature Lower Compartment, F	100	100	SM
Initial Pressure, psia	14.7	15.0	SM
Weight of Ice, lb.	2.45×10^6	2.1×10^6 lb.	85.7%
Ice Temperature, F	20	15	5F lower
Materials and Slabs			
Concrete Thermal Conductivity, BTU/(hrftF)	0.8	0.84	105%
Concrete Heat Capacity, BTU/lb	0.238	0.204	86%
Concrete Density, lb/ft^3	158	14.8	94%
Concrete Volumetric Heat Capacity, BTU/(ft^{3}F)	38.5	30.2	78.5%
Total Concrete Wall Volume, ft^3	267,185	282,636	SM
Total Steel Volume, ft^3	10,243	9,651	SM

Table A.18 (Continued)

	SARP[9]	IDCOR[11]	IDCOR/SARP % Data or Comment
Engineered Safety Systems			
Charging Pump			
Maximum Flow, gpm	1,100	2 pumps x 492	89%
Shut Off Pressure Setpoint, psig	2,530	1,885	--
ECC HPI Pump			
Maximum Flow, gpm	1,300	2 pumps x 650	SM
Shut Off Pressure Setpoint, psig	1,520	1,489	SM
ECC LPI Pump			
Maximum Flow, gpm	6,000	2 pumps x 6,063	202%
Shut Off Pressure Setpoint, psig	210	167	
Containment Spray Pump, gpm	4,750	2 pumps x 4,746	200%
Pressure Setpoint of Sprays, psia	--	17.51	--
ECR Heat Exchanger (HPI*LPI Pump Flow			
Primary Flow, lb/min	37,260	2 pumps x 55,305	2x148%
Secondary Flow, lb/min	41,400	2 pumps x 82,667	2x200%
Primary Inlet, F	137	Calculated (137F assumed)	--
Secondary Inlet, F	91	100	--
Capacity, BTU/hr For ΔT = 17F in primary circuit)	3.72×10^7	$2 \times 5.66 \times 10^7$	2x152%

Table A.18 (Continued)

	SARP[9]	IDCOR[11]	IDCOR/SARP % Data or Comment
Spray Heat Exchanger			
Primary Flow, lb/min	39,333	2 pumps x 39,223	2x100%
Secondary Flow, lb/min	49,910	2 pumps x 82,888	2x166%
Primary Inlet, F	146	Calculated (156F assumed)	--
Secondary Inlet, F	83	100	--
Capacity, BTU/hr (For ΔT = 40F in primary circuit)	9.43×10^{7}	$2\times9.41\times10^{7}$	2x100%
Recirculation Air Fans, gal air/min	--	299,200	--
ECC Storage and Injection Tanks			
UHI Tank: Water Content, lb	93,000	113,500	122%
Initial Pressure, psia	1326	1255	SM
Temperature, F	100	120	--
Accumulator Tank: Water Content, lb	230,000	4x65,100	113%
Initial Pressure, psia	600	415	--
Temperature, F	100	100	SM
RWST Tank: Water Content, lb	2.90×10^{6}	2.909×10^{6}	SM
Initial Pressure, psia	14.7	15.0	SM
Temperature, F	100	105	SM

Table A.18 (Continued)

	SARP[9]	IDCOR[11]	IDCOR/SARP % Data or Comment
Core Heatup Model			
Number of Radial Zones	10	7	SM
Number of Axial Zones	24	10	SM
Containment Break Area, ft^2			
S_2HF Accident (Drains Open)	7.0	0.1	1.4%
TMLB', S_2HF (Drains Blocked)	7.0	0.02	0.3%
Containment Failure Pressure, psia	60	65	108%

SM = Similar data.

U = Underknown for IDCOR case.

-- = Different data.

Table A.19 NRC/IDCOR Issues

Issue	Subject
1	Fission-Product Release Prior to Vessel Falure
2	Recirculation of Coolant in Reactor Vessel
3	Release Model of Control Rod Materials
4	Fission-Product and Aerosol Retention in the Primary System
5	In-Vessel H_2 Generation
6	Core Slump, Core Collapse, and Reactor Vessel Failure
7	Containment Failure due to In-Vessel Steam Explosions
8	Direct Heating of Containment
9	Ex-Vessel Fission-Product Release
10	Ex-Vessel Heat Transfer Model from Molten Core to Concrete
11	Revaporization of Fission Products from the Primary System
12	Fission-Product Deposition Model in Containment
13A	Suppression Pool Bypass (Pool Scrubbing)
13B	Retention of Fission Products in Ice Beds
14	Modeling of Emergency Response
15	Containment Performance
16	Secondary Containment Performance
17	Hydrogen Ignition and Burning
18	Essential Equipment Performance

Table A.20 Consequence Bins for IDCOR Baseline, Sequoyah

Bin	Defining Sequence	Time of Release (hrs)	Warning Time (hrs)	Conditional Consequences* Latent Cancer Fatalities	Whole-Body Population Dose (man-rem)
1	S_2HF-β (drains open)	5.3	0.5 117	140	2.0E+6
2	T_1B_3MLB'-δ	27	25	47	7.3E+5
3	β/W**	5.3	0.5	31	4.6E+5
4	S_2HF-δ (drains open)	9.8	8.5	13	2.0E+5
5	V	18	17	4.0	5.6E+4
6	NCF	NA	NA	0	0

*No early fatalities were calculated for these releases.

**Containment isolation failure with sprays available. Consequences based on estimated reduction of fission products due to spray scrubbing.

NA = Not Applicable and/or Available.

Table A.21 List of Symbols for Reactor Accidents

A	Intermediate to large LOCA.
B	Failure of electric power to ESF's.
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, T_1.
D	Failure of the emergency core cooling injection system.
F	Failure of the containment spray recirculation system.
F_1, F_2, F_3	Different variants of F.
H	Failure of the emergency core cooling recirculation 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.
S_1	A small LOCA with an equivalent diameter of about 2-6 inches.
S_2	A small LOCA with an equivalent diameter of 1/2-2 inches.
S_3	Small accident with an equivalent diameter 0.75 inch = pump seal LOCA.
SGTR	Steam generator tube rupture.
T	Transient event.
T_{23}	Transient event other than T_1 = transient from loss of offsite power.
U	Chemical and volume control system.
V	LPIS check valve failure.
W	Failure to remove residual core heat.
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.
θ	Water present in cavity at RPV failure time.