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Charles W. Elliott, Esq.
Brose & Poswistilo
1101 Building
11th & Northampton Streets
Easton, Pennsylvania 18042

In the Matter of
PHILADELPHIA ELECTRIC COMPANY
(Limerick Generating Station, Units 1 and 2)
Docket Nos. 50-352 and 50-353

Dear Mr. Elliott:

By letter dated March 22, 1984 we forwarded to you the monthly reports for November through February in connection with Contract NRC 03-83-092 and advised you that when a final report becomes available we would provide you with a copy. Accordingly, I am enclosing for your information "State of the Art of Reactor Containment Systems, Dominant Failure Modes, and Mitigation Opportunities."

Sincerely,

Joseph Rutberg
Assistant Chief Hearing Counsel

Enclosure: As stated

cc: See page 2

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cc w/enclosure:
Lawrence Brenner, Esq., Chairman
Dr. Peter A. Morris
Dr. Richard F. Cole
Troy B. Conner, Jr., Esq.
Dir. Pa. Emer. Mgmt. Agency
Martha W. Bush
Gregory Minor
Zori G. Ferkin
Mark J. Wetterhahn, Esq.
Atomic Safety and Licensing Board
Atomic Safety and Licensing Appeal Board Panel
Docketing and Service Section

cc w/o enclosure:
David Wersan
James Wiggins
Frank R. Romano
Joseph H. White III
Robert L. Anthony
Timothy R. S. Campbell, Director
Marvin I. Lewis
Angus Love, Esq.
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Sugarman and Denworth
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RDA-TR-127301-001

STATE OF THE ART OF REACTOR CONTAINMENT SYSTEMS, DOMINANT FAILURE MODES, AND MITIGATION OPPORTUNITIES

Final Report
January 1984

James N. Castle
Ivan Catton
James L. Dooley
R. Philip Hammond
William E. Kastenberg
David Swanson

R & D Associates
P.O. Box 9695
Marina del Rey, CA 90292

Prepared for
U.S. Nuclear Regulatory Commission

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The views expressed in this report are not necessarily those of the U.S. Nuclear Regulatory Commission.

ABSTRACT

The state of the art of nuclear reactor containment structures is described, and the five major types in the U.S. are compared. For each type, the report analyzes the dominant modes by which a severe accident could cause a containment to fail, the degree of public risk that would result if it did fail, and the feasibility of modifying the system to prevent such failures. The report is the first in a series of five related studies aimed at developing a coherent approach for assessing the practicality, cost and value/impact of a mitigation program for rendering a containment relatively resistant to severe core-melt accidents.

PREFACE

This report describes the first of a group of five studies which constitute a program known as Severe Accident Consequences Mitigation. It is being carried out under the direction of the Reactor Systems Branch, Division of Systems Integration, U.S. Nuclear Regulatory Commission (NRC). The five studies are entitled:

- Summary of State-of-the-Art Reactor Containment Systems, Dominant Failure Modes, and Mitigation Opportunities.
- Survey of the State of the Art in Mitigation Systems.
- Detailed Conceptual Design and Feasibility Study of Existing, New, or Improved Mitigation Systems.
- Value/Impact Analysis of Mitigation Systems/ Approaches.
- Development of a Risk Mitigation Strategy and Methodology for Licensing Activities and Related Decisionmaking.

The overall program will provide the NRC staff with designs, detailed cost data, and value rankings of mitigation systems for specific reactor containments. In addition, evaluation methods and regulatory approaches will be recommended for assessing mitigation options in terms of their value and impact for reducing public risk. Based on studies of actual plants under licensing review, on studies of generic phenomena, and on recent research findings, this program will define the technical, regulatory and economic feasibility of mitigation options, together with a possible strategy for their implementation. It must be stressed, however, that the risks presented in this report represent a "snapshot in time;" i.e., as the methods and data change with time (e.g., new source term information), the calculated risks will change. Hence, this report should be viewed as a preliminary reference document with the quantitative values subject to change. A final version will be published at a later date.

CONTENTS

CONTENTS (Continued)

CONTENTS (Concluded)

ILLUSTRATIONS

ILLUSTRATIONS (CONCLUDED)

TABLES

TABLES (CONTINUED)

TABLES (CONTINUED)

TABLES (CONCLUDED)

CHAPTER 1. INTRODUCTION

1.1 BACKGROUND OF THE STUDY

The public safety record of U.S. nuclear power plants has not been attained by eliminating all errors, failures and accidents, but by applying an important principle called "Defense in Depth." This means that every important function in the operation of the plant is backed up by an alternative system, or several systems, that can either assume the original function or handle the consequences of not doing so.

The key source of public risk in a nuclear power plant is the radioactive material in the fuel. This material is kept isolated from the public by three separate barriers, each of which contributes to the Defense-in-Depth concept. The primary confinement of the fuel is the cladding. For a water-cooled reactor, the principal means of insuring integrity of the cladding is keeping it cooled with liquid water. There are several redundant and independent means for doing this under a variety of normal and off-normal situations.

The second line of defense is provided by the reactor pressure vessel and the primary piping, which form a closed integral system capable of confining radioactive material even if the fuel cladding is partially damaged. The primary system is strongly protected against mechanical and seismic damage, and is designed and manufactured to a proven high standard of quality.

The final barrier, or third line of defense for the protection of the public, is the containment building--a pressure-tight structure that encloses the reactor vessel and the primary system and is capable of withstanding major failures of the first two barrier systems. The containment is a formidable bulwark designed to withstand a spectrum of conceivable threats arising from external causes (e.g., tornadoes, hurricanes, air crashes, and earthquakes) and from major failures within the reactor system. Several different versions of containment design have evolved with the development of nuclear power, each matched to a particular type of reactor by a particular vendor.

The combination of Defense in Depth for the primary reactor system and its containment building is intended to ensure that the plant poses no "undue risk to the health and safety of the public." Recent assessments of the possibilities of reducing this residual risk still further have produced some unexpected results. By far the most satisfying way to

reduce public risk is by prevention. In this context, the initial sequence of plant failures that ultimately might develop into a breach of the three protective barriers is interrupted or prevented. Analysis has shown, however, that the number of pathways or combinations of minor failures that could lead to major failures is very large, so that it is almost impossible to prove that all of them have been discovered and provided with a "fix." Indeed, the Three Mile Island (TMI) accident brought this point home very dramatically. In this case, the first barrier--the fuel cladding--was damaged. A valve stuck partially open formed a temporary failure of the second barrier, the primary system; and the operator's inadvertent removal of highly radioactive water through an emergency line to the auxiliary building permitted release of gaseous fission products through the "stack," thus penetrating the third barrier, which was, nevertheless, still structurally intact.

The complexity of the pathways to a plant failure has led to a possibility that the residual public risk might actually be increased by further attempts to backfit the reactor system, since the various changes could begin to interact adversely with existing systems. In this situation, attempts to further reduce public risk might be less equivocable if the third level of defense--the containment barrier--is improved. This barrier, though capable of handling various internal failures, was never designed to withstand internal events more serious than the so-called "design basis accident" (DBA), wherein the core and the primary system might be damaged but complete meltdown does not occur. Removing this limitation to include most accidents seems feasible, because studies show that current containment buildings are potentially capable of withstanding much more insult than was intended in their design.

If the containment barrier could be enhanced to completely contain almost any type of internal failure, two important advantages would accrue. First, the design of such facilities would tend to be more simple and more certain in function than further backfitting to the complex reactor system itself. This follows from the fact that although the pathways that could lead to a damaged reactor are numerous and complex, the number of end-states that result are much fewer in number and can be better characterized. Second, a higher level of confidence from a technical standpoint should be attainable for this type of protection. Such higher confidence would derive both from the fact that the containment and its function are relatively independent of the reactor system, and from the inherent differences between the art of containment design and the design of reactors. In an idealized sense, containments cannot fail to work if they are in

the right place at the right time and are made of the right materials, while the reactors themselves require instruments, sensors, mechanisms and operators all functioning in a highly coordinated manner.

Recent results obtained from probabilistic risk assessment (PRA) studies show that existing containment types, with minor modifications, are capable of protecting the public against accidents more severe than that for which they were intended. The DBA was chosen some years ago as a limit beyond which the residual public risk averted was deemed not to justify the level of cost involved to achieve it. This report, and the other activities of which it is a part, represents a portion of the NRC's reexamination of that earlier chosen limit. Provided here is an up-to-date examination of the state of the art of reactor containments and of the possibilities for augmenting their capabilities in the event of a severe accident. The emphasis is thus not upon prevention of an accident, however desirable that is, but upon mitigation of its consequences--specifically, those consequences that might endanger the health and safety of the public. Since the different types of reactor containments have different modes of failure and different possibilities for mitigation, each is addressed in a separate chapter.

1.2 REGULATORY BACKGROUND AND PROCEDURE

The NRC published the "Advanced Notice of Rulemaking" in the Federal Register on 2 October 1980 to serve notice that a long-term effort was to be initiated that would establish policy, goals, and requirements for core-melt accidents greater than the present DBA. It was very quickly discovered that the technological basis for decisionmaking was weak. This led to the NRC augmenting its Severe Accident Research Program (SARP). The nuclear industry responded to the advance notice of a possible new design basis by organizing and implementing the Industry Degraded Core Rulemaking (IDCOR) group.

The first step in the decisionmaking process will be to decide how safe nuclear power plants are and whether this is safe enough. This judgment will be made by judicious application of risk assessment, analysis, and engineering judgment. Once this is done, the question of whether or not something should be done can be answered. If risk reduction is deemed necessary, then the question of whether it is to be accomplished by prevention, mitigation, or administrative procedures must be addressed. The various research programs sponsored by the NRC (over 50 of these are relevant to the subject) and IDCOR are intended to show how best to achieve

risk reduction if it is desired. Separate efforts address the three methods of risk reduction. In this report and its sequels, the focus is on mitigation.

A number of complex technical issues must be addressed before decisionmaking on regulatory issues can be made without high uncertainty. At present, substantial uncertainties exist in both the phenomenology of a severe accident and in the public risk that it produces. Well engineered mitigation devices can remove much of the phenomenological uncertainty, but the benefit of doing so goes back to the risks inherent in the unmodified system. The net result is certainty in the mitigation function but large uncertainty in the actual risk reduction. Mitigation of core-melt consequences by design changes or by the addition of various devices has been and is the subject of a number of studies. A discussion of some of the studies sponsored by the NRC Division of Reactor Safety Research is given in Chapter 2 of this report.

The present study is one of several related NRC-sponsored studies. The IDCOR group, the Electric Power Research Institute (EPRI) and others have also sponsored studies on accident mitigation. This work will differ from the others in the depth to which the design aspect is carried. Details of other relevant work are reviewed as a part of Task 2 which is reported in a separate document.

1.3 DEFINITION OF A MITIGATION SYSTEM

For the purposes of this study, the NRC has defined severe accident mitigation as those actions, devices, or systems intended to reduce, ameliorate, or remove the consequences to the public of a severe accident wherein the core is degraded or melted. Accident prevention, on the other hand, will be the term applied to those activities related to controlling the condition of the reactor and fuel before the core is damaged or melted. As with any such definition, ambiguous cases will appear. The Three Mile Island accident, for example, is a borderline case. Changes in the reactor equipment, instruments, and valves to prevent a repetition of such an accident would be prevention, but correcting the means by which small amounts of radioactive gas were vented to the atmosphere could be called mitigation. Failure to scram following an anticipated transient (ATWS event) represents another ambiguous case for some reactor containments, in which a boiling water reactor might cause overpressure failure of the containment long before the core is damaged. In this particular case, the distinction has been resolved by allocating to prevention any measures to avert the initiation of the ATWS, and to mitigation the steps taken to reduce the consequences of a core

melt or overpressure that might follow later in the same accident. As an example, such mitigation might take the form of a self-closing pressure relief valve in the containment, capable of venting off enough noncontaminated steam during the event to prevent rupture. This valve would have reclosed before the reactor finally boiled dry, became subcritical, and melted. Hence, any subsequent release of radioactive materials would be contained.

As noted above, mitigation can include actions or procedures, devices, or combinations of components to form mitigation systems. In practice, a reactor meltdown accident would seldom pose only a single threat to the barriers protecting the public or the environment. There is the risk of overpressure failure of the containment due to steam, hydrogen formation, or concrete attack. There is the need to prevent escape of the core materials by melting through the bottom of the containment. Finally, there is the necessity of discharging outside the containment the heat generated by the contained core without permitting the escape of radioactive materials. Unless the mitigation system covers all the threats in a particular scenario, the amount of risk reduction will be less than optimum, and may not be cost effective. The actual risk averted will depend upon the dominant modes of containment failure, the plant arrangement, and the population density of the site; but in general, mitigation of a severe core-melt accident will require consideration of overpressure control, containment heat removal, and core retention.

For the purposes of this report then, a containment mitigation system will denote a cooperative combination of devices, subsystems, and components capable of dealing with the various modes of containment failure for a particular plant. The components of such a system will vary according to the threats deemed worthy of mitigation by value/impact analysis. Operator action can be a part of such a system, and it must be emphasized that in this report description of a piece of hardware suitable to perform a required mitigation function does not preclude other means of performing the same function. Operator action or modification of existing equipment can possibly perform as well as dedicated hardware in some cases and at lower cost.

At present, the human factor with planned intervention has recently been recognized to be an important possibility for accomplishing accident mitigation. Current activities include study of computerized operator aids for early diagnosis, improved instrumentation and display, alarm prioritization, extension of plant emergency procedures, and others. Operator training for such extreme conditions does not exist

at the present time. Even without planned intervention, it is difficult to assess what the human factor contribution might be. Present PRAs are based on judgment rather than on solidly based actuarial data. There is no doubt that a well-trained operator who understands the physical processes guiding the course of the accident could intervene successfully in some cases. Only after a great deal more study will we know how he should be trained and specifically what his procedures should be. It may be that the operator will be much more useful as part of a prevention scheme than a mitigation scheme.

1.4 PLAN OF THIS REPORT

This study of reactor containment capability and opportunities for enhancement is aimed at practical application in assisting the development of regulatory policy. Hence, it cannot be generic, but must be directed to specific types of containments. Electric utilities in the United States have a total of 139 reactor units built or committed; the containment structures housing 137 of these units fall into five major classes, each having many slight variations, as shown in Table 1-1, plus a few much smaller classes, some having only one member. To fulfill the regulatory purposes of this work, attention has been concentrated on the major classes, representing the large majority of the nuclear units. Table 1-2 lists salient characteristics of a representative plant for each of the five classes. Chapter 2 outlines the regulatory environment governing the study, the analytical tools and procedures that will be used to assess mitigation benefits, and a framework for determining mitigation requirements.

A full chapter of this report is then devoted to each of the five classes of containment, using a specific plant or plants for which a PRA report exists as the example for the class. Although variations exist among the members in any given class, these may be minor in their effect on the conclusions reached. Where possible, these variations and their consequences have been noted; some of the minor classes are also discussed as subclasses of the main group.

Within each main chapter the procedure is to describe first the containment system and the principal reactor and safety systems that interact with it. The dominant failure modes and associated risk probabilities are then discussed, together with the time scale over which hypothetical failure of the containment develops. The degree to which mitigation can potentially reduce the public risk is then explored, leading to a determination of the specific opportunities that exist in that containment class for mitigating specific

TABLE 1-1. MAJOR TYPES OF CONTAINMENTS FOR U.S. REACTORS

	Prestressed Cylinder + Dome	Reinforced Concrete + Dome	Other Concrete	Steel Sphere	Steel Cylinder, Domed Ends	Steel Dome, Concrete Base	Light Bulb + Torus	Total
Dry containment (PWR)	50	15	2	6	8			81
Ice suppression (PWR)		2				8		10
Mark I (BWR)			2				22	24
Mark II (BWR)			10					10
Mark III (BWR)		6				6		12

TABLE 1-2. TYPICAL CONTAINMENT PARAMETERS

Containment Type	Representative Plant	Free Volume (10^6 ft^3)	Primary Containment Design Pressure (psig)
1. Mark I	Peach Bottom	0.28	56
2. Mark II	Limerick	0.40	55
3. Mark III	Grand Gulf	1.7	15
4. Ice condenser	Sequoyah	1.2	11
5. Dry containment			
a. Prestressed concrete	TMI-1	2.0	55
b. Reinforced concrete	Comanche Peak	2.5	50
c. Steel cylinder	St. Lucie	2.5	40
d. Steel sphere	Yellow Creek	3.6	45
e. Subatmospheric	Surry	1.8	45

accident conditions or failure modes, and the special requirements to be met. Finally the kinds of mitigation hardware that might meet the requirements are described briefly. A companion report in this same series, "Survey of the State of the Art in Mitigation Systems," provides a more comprehensive description of available hardware.

Following the five chapters on specific types of containment, the overall findings are discussed and compared to provide some perspective on the regulatory utility of mitigation activities and installations. Throughout the report the attempt has been made to keep the thread of the discussion clear and readable, yet to provide full documentation through adequate references and the use of appendices as appropriate.

The nomenclature used in each chapter for describing dominant accident sequences and containment failure modes, as well as the grouping of these sequences and modes into radioactive release categories and/or containment failure class, differs from chapter to chapter. Rather than attempting to use a uniform nomenclature throughout, each chapter uses the nomenclature found in the literature for each containment type, which in itself is variable. In this way the reader familiar with the references will also recognize the meaning of the symbols used here.

Table 1-3 presents a summary of the nomenclature used for each containment failure mode as presented in each chapter. Table 1-4 summarizes the nomenclature used for containment failure class as well as for radioactive release categories in each chapter. Note that the grouping of containment failures into classes (as opposed to modes) is not done for all five containment types. For the BWR Mark I containment system, the dominant accident sequences are grouped by release category; each release category has been defined in the Reactor Safety Study (RSS) (NRC, 1975) and corresponds to a particular containment failure mode. For the BWR Mark II containment, the dominant accident sequences are grouped into containment failure classes. Release categories are derived for the various containment failure modes, and conditional probabilities of each mode link the containment class with the release category. The BWR Mark III containments are treated the same way as the Mark I containments except that the release categories correspond to different containment failure modes. Note that the release category describes the nature of the radioactive release, so that consequences can be determined, uncoupled from the frequency of that release. The PWR large dry containment is characterized in terms of the containment failure classes defined

TABLE 1-3. NOMENCLATURE FOR CONTAINMENT FAILURE MODES

Containment Failure Mode	BWR			PWR	
	Mark I	Mark II	Mark III	Large Dry	Ice Condenser
In-vessel steam explosion	α	α	α	α	α
Ex-vessel steam explosion	β	β	-	α	α
Overpressure: release to buildings	γ	-	γ	-	-
Overpressure: release to atmosphere	γ'	-	δ	-	-
Overpressure: drywell	-	γ	-	-	-
Overpressure: wetwell	-	γ'	-	-	-
Overpressure: wetwell + loss of suppression pool	-	γ''	-	-	-
Early overpressure	-	-	-	δ_1	δ
Late overpressure	-	-	-	δ_2	δ or δ_2
Containment isolation: drywell	δ	-	-	-	-
Containment isolation: wetwell	ε	-	-	-	-
Containment isolation	-	-	-	β	β
Reactor building isolation	η	-	-	-	-
Hydrogen burn	-	μ		γ	γ
Hydrogen detonation	-	μ'		-	-
Containment bypass	-	-	-	ν	ν
Containment leakage (large)	ξ	δ_ξ	β	-	-
Small leaks	-	δ	-	-	-
Small leaks: SGTS* failure	θ	δ_τ	-	-	-
Basemat melt-through	-	-	-	ε	ε

*SGTS: Standby Gas Treatment System

TABLE 1-4. CONTAINMENT FAILURE CLASS AND RELEASE CATEGORY NOMENCLATURE FOR EACH CONTAINMENT TYPE

BWR Mark I

Release Categories

BWR 1: In-vessel steam explosion (α)

BWR 2: Overpressure; release to atmosphere (γ')

BWR 3: Overpressure; release to reactor building (γ)

BWR 4: Containment isolation failure (δ)

BWR Mark II

Containment Failure Class

Class I: Core melt prior to containment failure

Class II: Containment failure prior to core melt

Class III: Core melt prior to containment failure (ATWS)

Class IV: Containment failure prior to core melt (ATWS)

Release Categories

OPREL: Overpressure release: all δ, γ and μ modes

OXRE: Oxidation release: α and β modes

C4γ: Overpressure ATWS and drywell

C4γ': Overpressure ATWS; wetwell

C4γ'': Overpressure ATWS; wetwell, no pool

BWR Mark III

Release Categories

BWR 1: In-vessel steam explosion (α)

BWR 2: Overpressure, release to atmosphere (δ)

BWR 3: Overpressure, release to reactor building (γ)

BWR 4: Containment leakage, large (β)

TABLE 1-4. CONTAINMENT FAILURE CLASS AND RELEASE CATEGORY NOMENCLATURE FOR EACH CONTAINMENT TYPE (CONCLUDED)

PWR Large Dry Containment

Containment Failure Class

Class 1: Containment by-pass (V)

Class 2: E, L

Class 3: EF

Class 4: EFC, LFC

Class 5: LF

Class 6: E, external initiators only

Nomenclature

E = Early core melt with no containment heat removal

L = Late core melt with no containment heat removal

EFC = Early core melt with containment heat removal

LF = Late core-melt sequences with only containment fan coolers operating

LFC = Late core melt with containment heat removal

EF = Early core-melt sequences with only containment fan coolers operating

Note: Containment Heat Removal Systems (CHRS) are both containment spray and fan coolers.

PWR - Ice Condenser Containment

Release Categories

PWR 1: In-vessel steam explosions (α)

PWR 2: Hydrogen burn, before vessel melt-through (γ)

PWR 3: Slow overpressure, after vessel melt-through (δ or δ_2)

PWR 4: Hydrogen burn, after vessel melt-through (γ)

PWR 5: Hydrogen burn, after vessel melt-through (γ)

PWR 6: Basemat penetration (ε)

by Brookhaven National Laboratory (Pratt, 1983). Consequences are given for each containment failure class. For the PWR ice condenser containment, the dominant accident sequences are grouped according to the PWR release categories defined in the RSS. These also correspond to a particular containment failure mode.

CHAPTER 2. APPROACH AND METHODS

This chapter describes the overall NRC activities on severe accident mitigation and discusses the benefits that can be derived from installing an accident mitigation system for a given reactor. The method used for translating this potential benefit into a set of specific functional requirements is then outlined.

2.1 MITIGATION-RELATED ACTIVITIES

Relative to other aspects of the degraded core accident, mitigation has received very little study so far. Severe Accident Sequence Assessment (SASA) studies on boiling water reactors (BWRs) at Oak Ridge National Laboratory (ORNL) (Cook, 1983; Condon, 1983) have yielded a great deal of insight but were not done with mitigation in mind. Similar studies at the Idaho National Engineering Laboratory (INEL), the Los Alamos National Laboratory (LANL), and the Sandia National Laboratories (SNL) on pressurized water reactors (PWRs) have been initiated (see Table 2-1). As a result of the Three Mile Island-Unit 2 (TMI-2) accident, considerable effort has also

TABLE 2-1. NRC STUDIES OF DESIGN CHANGES TO MITIGATE SEVERE ACCIDENTS

Title	Laboratory	NRC Fin. No.
SARRP	SNL	A1322
Development and Analysis of Vent-Filtered Containment	SNL	A1220
Technology for Core-Melt Retention Concept Assessment	SNL	A1247
Hydrogen Combustion Preventive and Mitigative Schemes	SNL	A1336
SASA	SNL	A1258
SASA	INEL	A6354
SASA	ORNL	B0452
SASA	LANL	A7228
Effectiveness of LWR ESF Systems Under Severe Accident Conditions	PNL	B2444
Conceptual Design of Alternative Core Melt Mitigation Systems for Ice Condenser Containment	INEL	A6351

gone into the study of mitigating the damaging effects of hydrogen combustion or detonation (Nelson and Berman, 1983). A study of proposed improvements to BWR pressure suppression and PWR ice-condenser containments carried out by Levy (1981) considered filtered venting and augmented decay heat removal (DHR) among other mitigation steps. A companion study for large dry containments sponsored by EPRI was never published. An INEL study by Reilly (1982) considered a pebble bed, late venting, water management, and a gas combustor as ways for reducing the consequences of a core melt accident. A study by Swanson, Castle and Catton (1983) proposed a flooded pebble bed core-retention system with containment heat removal that could be retrofitted into existing reactors. Work by Kastenberg and Catton (1983) examined heat pipe systems for containment heat removal. Hammond (1982) studied the possibility of installing a water-cooled crucible in a tunnel under the basemat. Work not yet published by Benjamin at SNL will consider a passive hydrogen control, a cavity flooding system, and a long-term containment cooling system as well as specific hardware additions and procedural changes. Past work relevant to the present study will be briefly described here.

The primary emphasis of the hydrogen damage mitigation system studies at SNL (Nelson and Berman, 1983) addressed the following schemes:

1. Deliberate ignition.
2. Deliberate ignition with pressure suppression.
3. Deliberate flaring.
4. Catalytic combustion.
5. Partial pre-accident oxygen reduction.
6. Full inerting.

Deliberate ignition systems for ice condensers (Sequoyah) and BWR Mark IIIs (Grand Gulf) have been proposed, evaluated, and installed. The ability of igniters (glow-plugs and spark coils) to function in spray and steam environments was given preliminary study for Sequoyah and is undergoing continuing experimental study at SNL. Work on the other items listed above is underway with formal reports expected soon.

The study by Levy (1981) presented the mitigation concepts in terms of functional needs for both the BWR Mark III and the PWR ice containment, i.e., hydrogen control, filter/vent, and reduction of noncondensible gases. An estimate was given for the risk reductions associated with various improvements to a BWR, and an argument was made that comparable benefits could be achieved for PWR ice containment.

The authors considered independent heat removal for loss of long-term cooling, improved standby liquid control for failure to shut down the reactor, atmospheric venting of relatively clean steam from a scram failure, pre- and post-inerting as well as burning for hydrogen control, and filtered vent for containment overpressure control. They concluded that benefits resulted from long-term cooling and controls for failure to scram, but not from the filtered vent.

Reilly (1982) selected the Sequoyah-1 plant for his study of core-melt accident mitigation for a PWR with an ice condenser containment. Results of the SNL study were used to identify the mitigation systems that would yield the greatest severe accident risk reduction. It was found that up to two orders of magnitude improvement could be achieved provided that uncontrolled hydrogen burning was prevented, that the interfacing valve sequences were controlled, that spray recirculation was maintained, and that the common mode loss of emergency core cooling was prevented. Three systems were selected for controlling basemat attack: a water management system to control the amount of water below the vessel, a pebble bed in the cavity, and a system to vent the containment upon excessive pressure. The three systems appeared feasible and relatively low in cost (under $500,000).

SNL is conducting the Severe Accident Risk Reduction Program (SARRP) as part of the NRC Severe Accident Research Plan (SARP). The intent of this program is to provide the NRC with a sound technical basis for severe accident decisionmaking. To achieve this intent, the SARRP has the following basic objectives:

1. Incorporate insights gained from severe accident research toward a rebaselining of reactor risks and of the overall uncertainty.

2. Investigate the use of generic plant categories as a means for generalizing the results of plant-specific risk analyses.

3. Evaluate the benefits and costs of proposed new safety features designed to reduce the frequencies and/or consequences of severe accidents.

The first objective (incorporation of insights) is being achieved through interfaces with ongoing programs and task forces, such as the Accident Sequence Evaluation Program (ASEP), the SARRP Phenomena Assessment task force, the Quantitative Uncertainty Estimation for the Source Term

(QUEST), and the NRC task forces on containment loading, containment response, and fission-product source terms.

The second objective (generalization of results) involves a cooperative effort with the ASEP program to define generic plant categories and to evaluate their risk. ASEP is deriving these plant categories by evaluating the sensitivity of risk to variations in system design. SARRP is then evaluating the sensitivity of risk to variations in containment design and siting factors.

For the third objective (cost-benefit analysis), the risk reduction benefits of a variety of safety options are being compared to the costs of implementation. The safety options under consideration include both systems that prevent the occurrence of core melting and systems that mitigate the consequences.

This and other ongoing programs are being coordinated with the severe accident mitigation studies of which this report is the first output. Later reports in this series will reflect this exchange of ideas and information to a greater extent.

2.2 MEASURING THE POTENTIAL BENEFITS OF MITIGATION SYSTEMS

The objective of the containment mitigation systems described in this report is to reduce, eliminate, or ameliorate the consequences of a severe accident that would otherwise lead to the release of radioactive materials to the environment. The measure of the potential benefits of these systems is usually given in terms of risk reduction or risk averted. In this context, it is convenient to express risk as

$$R^i = \sum_{j=1}^{J} \sum_{k=1}^{k} f_j \, P_{jk} \, C_{ki} \qquad (1)$$

where f_j = the frequency of the jth accident sequence leading to core melt,

P_{jk} = the conditional probability of the kth containment failure mode for the jth accident sequence, and

C_{ki} = the ith consequence of interest for the kth containment failure mode.

In most PRAs, several consequences are considered: man-rem, acute (early) fatalities, latent (cancer) fatalities, off-site property damage in dollars, and illness (usually thyroid cancer). Because of the differences in these consequences, R^i is not aggregated; i.e., one does not sum over i. The benefit or risk averted is determined by evaluating Equation (1) with (w) and without (w/o) mitigation, as

$$\Delta R^i = R^i_{w/o} - R^i_w \qquad (2)$$

In this report, the focus is on mitigation systems intended to improve containment performance for severe accidents involving core melt. Hence, it is desired to reduce or eliminate P_{jk} for selected containment failure modes. If one provides a perfect mitigation system for the kth containment failure mode, thereby eliminating its consequences, the benefit or risk averted reduces to

$$\Delta R^i_k = \sum_{j=1}^{J} f_j \, P_{jk} \, C_{ki} \qquad (3)$$

where f_j, P_{jk}, C_{ki} are the values without mitigation.

Note that for some reactor power plants, containment mitigation may also change the frequency of a core-melt sequence. For example, prevention of containment failure may ensure core integrity by preventing pump failure during a recirculation pump injection mode. This effect is ignored in this study, thus tending to introduce some conservatism into the initial calculated risk reductions. Effects of this type can be incorporated in a more detailed study.

Potential benefits of mitigation can also be measured by a risk reduction factor ($R^i_{w/o}/R^i_w$) for each consequence i. This formulation is useful in determining the relative importance of mitigating various containment failure modes for a given reactor plant, based on a single PRA.

In this report, existing PRAs are utilized to estimate potential risk reduction in terms of man-rem, acute (early) fatalities, and latent (cancer) fatalities for each mode of failure. Where more or less complete information is given (including the effects of external events) and where extensive review has been carried out, an attempt is also made to

monetize the risk according to the NRC-proposed cost effectiveness criterion of $1000/man-rem averted (NRC, 1982). An estimate of acceptable costs of mitigation can then be obtained by summing over the dominant failure modes. As discussed below, the act of intervening by prohibiting one mode of failure transfers the risks and hence benefit to another mode.

For those PRAs which do not include site-specific effects or external initiators, and/or have not undergone extensive review, risk reduction factors are presented as discussed above. It should also be mentioned that only those PRAs using the RSS methodology and source terms are utilized to illustrate potential risk reduction. This was done to provide a more consistent basis for comparison. PRAs which utilize new source term information such as that for GESSAR II are still under review.

The results presented here are only intended to show where mitigation opportunities exist for a given containment. They are not intended for regulatory decisionmaking. At a later stage of the project, various value/impact measures will be developed which, along with a normalization of the various PRAs, will provide a consistent basis for comparison.

Although the overall approach used here provides a useful basis for initiating the development of a regulatory policy, the reader should be aware of important limitations in the results presented. Three separate factors enter into any conclusions of mitigation feasibility: (1) how could the containment fail, (2) what remedies are feasible, and (3) what would the benefits be? Uniformly applied methods have been developed to answer the first two questions in a way that is consistent across all the five containment types, but the benefit assessment, in terms of risk averted, is based on PRAs that clearly lack such consistency. The PRAs cited in this report were done by different groups, using different procedures, different assumptions, and different levels of plant detail. In some cases, the effect of external initiators was included; in others not. Some PRAs have been reviewed extensively, and others lack such review. It is observable that the PRAs indicating the highest level of residual risk are those that have been reviewed most thoroughly. One would also expect that the most recent containment designs would represent the lowest residual risk, yet this is not always apparent from the PRAs. It should be stressed then that the risks presented here should be viewed as "snapshots in time;" i.e., as the methodology matures and new source term data become available, these calculated risks will change.

Although no attempt has been made in the numerical results shown to normalize these varying risk results across the different containment types, it is clear that further effort to accomplish this would be productive. The results given here would be readily correctible if standardized risk data could be obtained. In the comments and summary for each chapter, some note is taken of the probable effect of further PRA reviews and the reduction of uncertainty.

All the work in the field of accident sequences and containment failure analysis has been done to assess the residual risk from the as-built system. Thus, when a sequence terminates in a containment failure, the resulting public dose is estimated and the analysis ends. A different branch of the sequence would have a different termination point and a different risk result. Each branch choice excludes all others so that the probabilities can be properly summed to unity.

When mitigation steps are taken, the outcomes are significantly different: each branch that would terminate in a containment failure is now instead converted into a new sequence, wherein a severe accident is underway but does not terminate. The containment failure that formerly would have occurred has been forestalled, but the radioactive material is still present and some other mode of failure now becomes possible, with an entirely different probability than it exhibited as an independent mode.

Thus, the situation might arise wherein a dominant sequence of relatively high probability and high risk might be mitigated by forestalling an early overpressure failure, for example. But preventing this outcome does not interrupt the accident--the system cannot go back and follow another branch having a different outcome and probability. Instead, the accident in progress may now lead to a different mode of failure, such as a containment melt-through, which has the probability of the original failure (unless it in turn is averted), times the probability of any further adverse branches in the continued chain. The original probability of this mode of failure must then be added independently.

Although the overall probability still sums to unity, the benefits tend to be cumulative rather than independent. The attendant benefits derived from partial mitigation followed by such a cumulative failure are not available at present; they would have to be estimated anew. In this report such calculations are not attempted. We report the risk averted by applying mitigation to each sequence and take their sum. This requires that the actual mitigation steps undertaken be

sufficient to provide full control of all the dominant accident outcomes. This procedure permits a positive, rather than an inconclusive result to the study. It is essential that the reader understand this purpose.

2.3 THE PRINCIPLE OF CUMULATIVE FUNCTIONAL REQUIREMENTS

A degraded core or core-melt accident in a nuclear power plant carries a risk to the health and safety of the public because such accidents, although rare, initiate processes having a high probability of causing containment failure with consequent escape of radioactive materials. To eliminate such consequences, an essential requirement is that the final stage--containment failure--does not occur. However, for any given type of containment, more than one mode of failure is possible, depending upon the particular accident end-state that preceded it. The accident end-state condition is the name given to the situation within the reactor and the rest of the plant when accident prevention measures have all failed, damage to the core has occurred, and damage to the reactor vessel and ultimately to the containment is underway.

Each separate containment failure mode may represent the culmination of any of several accident end-states, and each end-state, in turn, represents the result of a much larger set of detailed accident sequences. The probability of any one particular accident sequence occurring may be very small, but the aggregate risk represented by a given failure mode may be substantial. Though only one accident sequence can occur in a real case, it is impossible to predict which one. To reduce the risk, one must assume, in effect, that all possible sequences have occurred simultaneously. Similarly, all the end-states leading to significant failure modes must be accounted for in the mitigation scheme. Thus, the governing requirements are established not by individual sets of failures, but by the envelope of them all.

For example, if it is found that loss of all site electric power is a dominant contributor to a given containment failure mode, then none of the mitigation devices used should depend upon site power. On the other hand, if none of the dominant sequences includes a failure to isolate the containment, then it is appropriate to assume that penetrations are properly closed in all mitigation conditions. The net result is that the requirements for mitigation are cumulative. Thus, for any given type of containment, the functions that any proposed mitigation design must fulfill will be derived from a combination of the significant failure modes.

Although this might seem an overly conservative approach, it becomes essential one if mitigation efforts are to be credited with a real reduction in risk. In practice, it is found that many of the requirements are redundant (being the same for several accident sequences) and that certain mitigation steps can fulfill several requirements at once. In addition, the incremental costs for overall mitigation tend to be lessened through use of common facilities and existing structures. The final result is that, after analysis of the dominant accident sequences and the end-state conditions thereby created for a particular type of containment, one can compile a set of "cumulative functional mitigation requirements." Such a list would comprise the necessary functions that a mitigation system must perform and the conditions under which it must operate.

It is important to distinguish between the "function" and the device that performs it. For example, to reduce the risk of containment overpressure failure from gases generated by concrete attack, both a filtered vent and a core catcher that prevents concrete attack would each perform the same function in a different way. The choice between such options would depend upon other functions that each device could perform; which one fitted best into an overall, minimum-cost system; and which one averted a larger risk. In the following chapters, the opportunities for accident mitigation are given for each containment type in terms of these functional requirements. From these requirements is derived a set of possible hardware choices that could fulfill them. This procedure is followed for each of the five major types of containment.

In implementing a program of mitigation for a particular plant, these hardware choices would be selected according to a value/impact analysis.

CHAPTER 3. BWR MARK II CONTAINMENT

The Mark II containment design was developed by General Electric (GE) for BWRs as a replacement for the Mark I containment design. Appendix D shows that ten U.S. plants fall into this category, with the earliest operational unit--La Salle I at Seneca, Illinois--having been granted an operating license in April 1982. Approximately half of the ten plants are still in the planning or construction stage and have not been granted operating licenses by the NRC. Power levels range from 810 to 1100 MWe. The Mark II design is of the pressure-suppression type and postdates the older and smaller Mark I BWR plant configuration.

The Mark II is an over-under pressure-suppression configuration (see Figure 3-1). The suppression pool--or wetwell--is located below the truncated cone-shaped drywell containing the reactor vessel. The suppression pool is used to accommodate safety relief valve (SRV) actuation after a turbine trip because the BWR cannot use an atmospheric dump. Suppression pool water is used to condense steam released during an accident and, in doing so, prevents an unacceptably large pressure in the containment. Pressure suppression with a wet well can be accomplished in a relatively small volume compared to that achieved with a dry containment which relies primarily on volume alone to accommodate major accidental steam releases. The large energy content of a BWR in the form of steam and water (compared to a PWR with a similar power rating) would require a very large and, therefore, expensive dry containment. Other benefits of the pressure-suppression containment are as follows:

1. Fission products can be scrubbed from steam and other gases bubbling through the suppression pool during serious accidents.

2. The reduction in containment pressure by condensation action lowers the leakage rate through the containment.

3. The suppression pool serves as a large heat sink. For example, 140,000 ft^3 of water--a typical amount--can absorb 100 MW-hr through a temperature rise of 40°F.

Sections 3.1 describes the general characteristics of the Mark II reactor system and its Engineered Safety Features. Attention focuses on the containment system and other safety designs that support containment integrity.

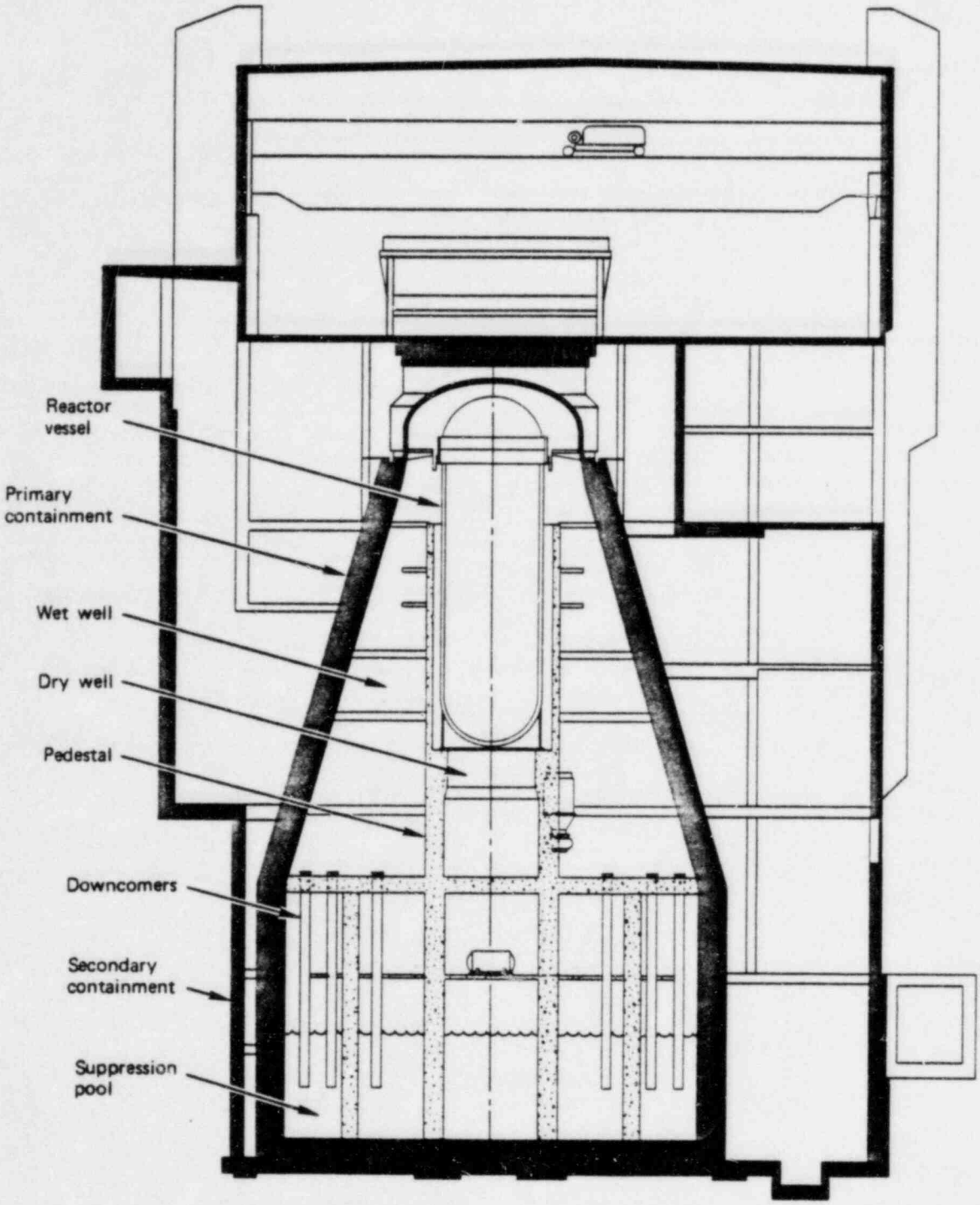


Figure 3-1. Mark II primary and secondary containments.

The serious accident sequences capable of challenging the containment are next identified in Section 3.2. Containment challenges can be created by Class 9 accidents which are, by definition, beyond the "Design Basis." The information sources for accident identification are published PRAs. Two PRAs are available: one each for the Shoreham and Limerick I plants. The important information drawn from the PRA is concerned primarily with the type of accident, the mode of radioactivity released from the containment, the quantity of radioactivity released, and the assumptions/limitations in the PRA analysis itself.

Section 3.3 includes an assessment of the theoretical extent to which mitigation systems may reduce accident risk. Risk is described in terms of man-rem, early fatalities, and latent fatalities. The value of any reduction in risk is quantified by assuming that each reduction in one man-rem is worth $1000. Man-rem are calculated out to 50 mi. Risk averted as well as a risk reduction factor is determined where appropriate. Mitigation systems are suggested that have the potential for reducing the quantity of radioactivity released in the unlikely event of a severe accident (Section 3.4). Recommendations are made in Section 3.5 for further study of those mitigation systems with a particularly high risk reduction value.

3.1 DESCRIPTION OF THE MARK II CONTAINMENT AND RELATED SAFETY SYSTEMS

This section of Chapter 3 provides a description of the Mark II plant environment into which severe accident mitigation systems may be fitted. The emphasis in the discussion is on the nuclear reactor itself, the reactor internals, the emergency core cooling systems (ECCSs), the primary and containment, and the Engineered Safety Features associated with the primary containment function. Items in the last category include the containment isolation system, the long-term containment heat removal system, and the combustible gas control equipment.

A severe accident mitigation system is likely to depend--at least in part--on a fully functioning primary containment. Therefore, the containment dimensions, design pressure, construction materials, internal heat sinks, and leakage rate are highlighted in the discussion. The complementary systems needed to maintain containment integrity are also of interest. Severe accident mitigation will probably require containment isolation, heat removal, and combustible gas control. The existing plant Engineered Safety Features that are designed to accomplish these same functions under DBA conditions are included in the discussion of this section.

Reference is made to Appendixes A and B for a review of the plant safety system strategy and DBA set.

3.1.1 Reactor and Primary System

The Mark II plants contain BWRs supplied by GE. The nuclear system consists of a single cycle, forced circulation reactor producing steam for direct use in a steam turbine. A conceptual drawing of the reactor is shown in Figure 3-2. To date, basically six models of the BWR have been sold by GE. Table 3-1 shows that models BWR/4 and BWR/5, introduced in 1966 and 1969 respectively, are used in the ten Mark II plants planned or in operation in the United States. The rated thermal output of the ten reactors ranges from 2436 to 3323 MWt, with corresponding steam production rates ranging up to 14,000,000 lb/hr. The primary system consists of the vessel, steam piping, turbine, condenser, condensate and feedwater pumps, feedwater piping, recirculating pumps, and pressure relief equipment.

The reactor vessel contains the nuclear core, steam separators and dryers, jet pumps, instrumentation, feedwater distribution lines, and other components. It is designed for a pressure of approximately 1250 psig and operates normally at about 1020 psia in the upper steam space. The vessel is made of carbon steel with an internal stainless steel cladding. It has an internal diameter of roughly 250 inches and is about 850 inches high. The wall thickness is

TABLE 3-1. GE BWR NUCLEAR REACTOR MODELS

Model	Year Introduced	Characteristic Plant	Comments
BWR/1	1955	Big Rock Point	Dry containment; initial commercial BWR
BWR/2	1963	Oyster Creek	Mark I containment; first turnkey plant
BWR/3	1965	Dresden 2	Mark I containment; first jet pump application; improved ECCS, spray and flood systems
BWR/4	1966	Browns Ferry	Mark I and II containment; increased power density
BWR/5	1969	Zimmer	Mark II containment; improved safeguards
BWR/6	1972	Grand Gulf	Mark III containment; increased ouput; improved ECCS performance; 8 X 8 fuel bundle

Source: Spencer, 1982

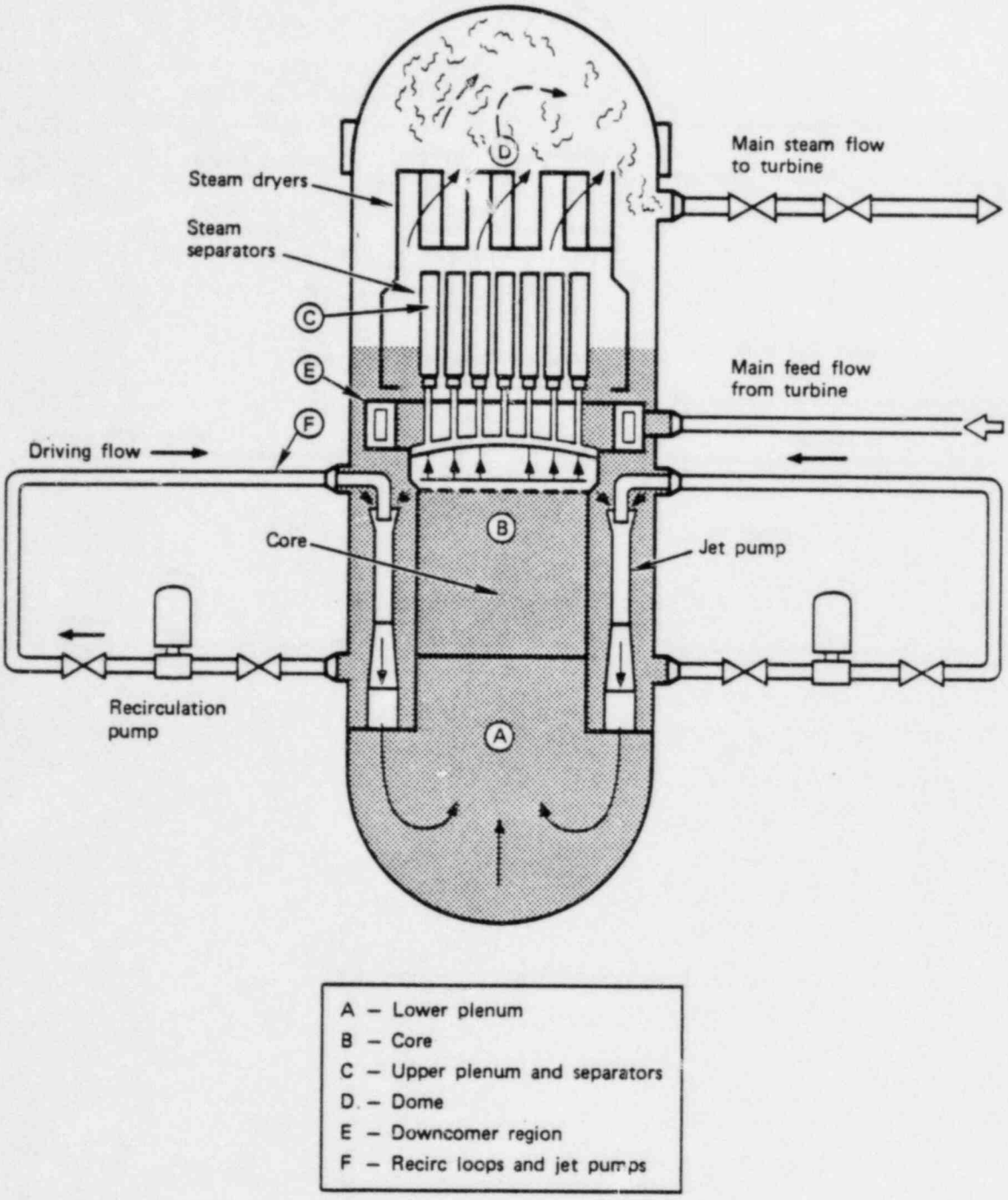


Figure 3-2. BWR reactor vessel internals.
Source: Spencer 1982

6 to 7 inches. Connections to the vessel include the steam lines, coolant recirculation lines, feedwater lines, control rod drive housings, and standby cooling lines. The control rods penetrate the vessel at the lower head. The reactor vessel is supported at its lower head by a circumferential skirt resting on the containment pedestal (see Figure 3-1). Cooling air passes upward along the vessel in the space between the vessel and the shield wall. This air enters through ducts in the pedestal wall below the vessel and it exits at the open top of the shield wall.

Slightly enriched uranium dioxide pellets sealed in Zircaloy tubes constitute the reactor fuel. The quantity of uranium oxide is 250,000 to 350,000 lb whereas the cladding amounts to 110,000 to 150,000 lb. The core itself ranges in diameter from 160 to 190 inches depending on the plant and has a height of about 150 inches (see Table 3-2). Gross reactor core control is achieved by use of moveable, bottom entry control rods. The control rod drives, support brackets, and piping occupy the pedestal region immediately below the reactor vessel.

TABLE 3-2. MARK II PLANT NUCLEAR SYSTEM
LIMERICK PLANT
(BWR/4)

Related power (MWt)	3293
Steam flowrate (lb/hr)	14.1×10^6
Number of fuel assemblies	764
Core UO_2 weight (tons)	~164
Core Zirconium weight (tons)	~48
Core height - active (in)	150
Core diameter (in)	187
Number of control rods	185
Number of jet pumps	20
Core spray system flowrate (gpm/loop)	6250 @ 122 psid (2 loops)
HPCI flowrate (gpm) (1)	5000
LPCI flowrate (gpm/pump)	10,000 (2 pumps)
RCIC flowrate (gpm) (1)	616 @ 1120 psid
RHR heat exchanger duty (Btu/hr/exchanger)	70×10^6 (4 - exchangers)

Steam is delivered by the reactor to the turbine-generator via four 24-inch diameter steam lines. Dual main steam line isolation valves are located on each of these lines where they pass through the primary containment wall and into the reactor building. The design temperature at the turbine inlet valve is about 560°F. The double-ended guillotine break of one of the steam lines is the worst-case loss of coolant accident (LOCA) included in the DBA set.

The reactor recirculation system uses two motor-driven (6000 to 7000 hp each) pumps to move coolant through the reactor. Valves on the recirculation lines can be used to control the reactor's power level through the effect of coolant flowrate on the void fraction in the core region.

Molten core debris is likely to move downward through the reactor's core support plate in the event of an extensive core-melt accident. Between the support plate and the lower vessel head is a forest of control rod guide tubes. Figure 3-3 shows two of the control rods out of a set that totals 90 to 185 individual units depending on the plant of interest. The guide tubes represent a weak link in the primary containment pressure boundary when subject to thermal attack. It is possible that thermally induced failure of the relatively thin guide tubes (see Figure 3-4) could release hot core debris into the containment well before the vessel head itself is breached.

A number of Engineered Safety Features serve to protect the integrity of the pressure boundary and the fuel rod cladding barrier of the primary reactor system. The first of these--the reactor protection system (RPS)--initiates rapid shutdown of the reactor (scram) in the event of abnormal operating conditions. It acts quickly to prevent cladding damage due to excessive temperatures (temperatures above 2200°F are potentially harmful to cladding integrity). Overpressure protection of the primary system is provided by a relief system consisting of multiple relief and safety valves mounted on the main steam lines. Steam discharged from these valves is directed to the pressure-suppression pool for cooling and condensation.

The reactor core isolation cooling (RCIC) system is another Engineered Safety Feature. It fulfills the important function of providing make-up water to the vessel whenever the vessel is isolated from the normal feedwater system. Water injection by the RCIC system is powered by steam drawn from the main steam lines. In addition to the RCIC system, four ECCSs are provided to help assure that the fuel cladding temperature remains within acceptable limits during accidents and other abnormal conditions. These systems are

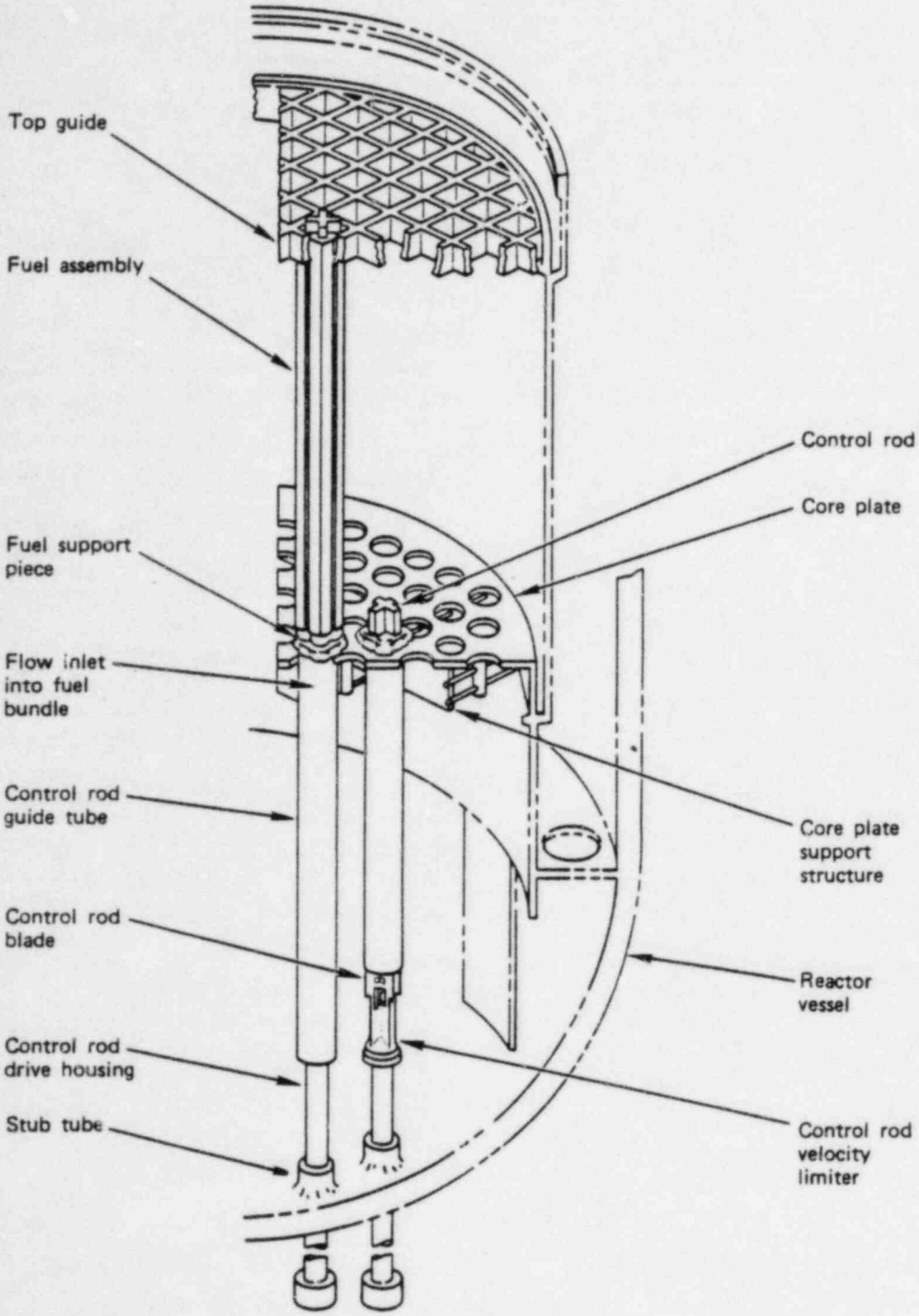


Figure 3-3. Control rod configuration in the BWR.
Source: Spencer 1982

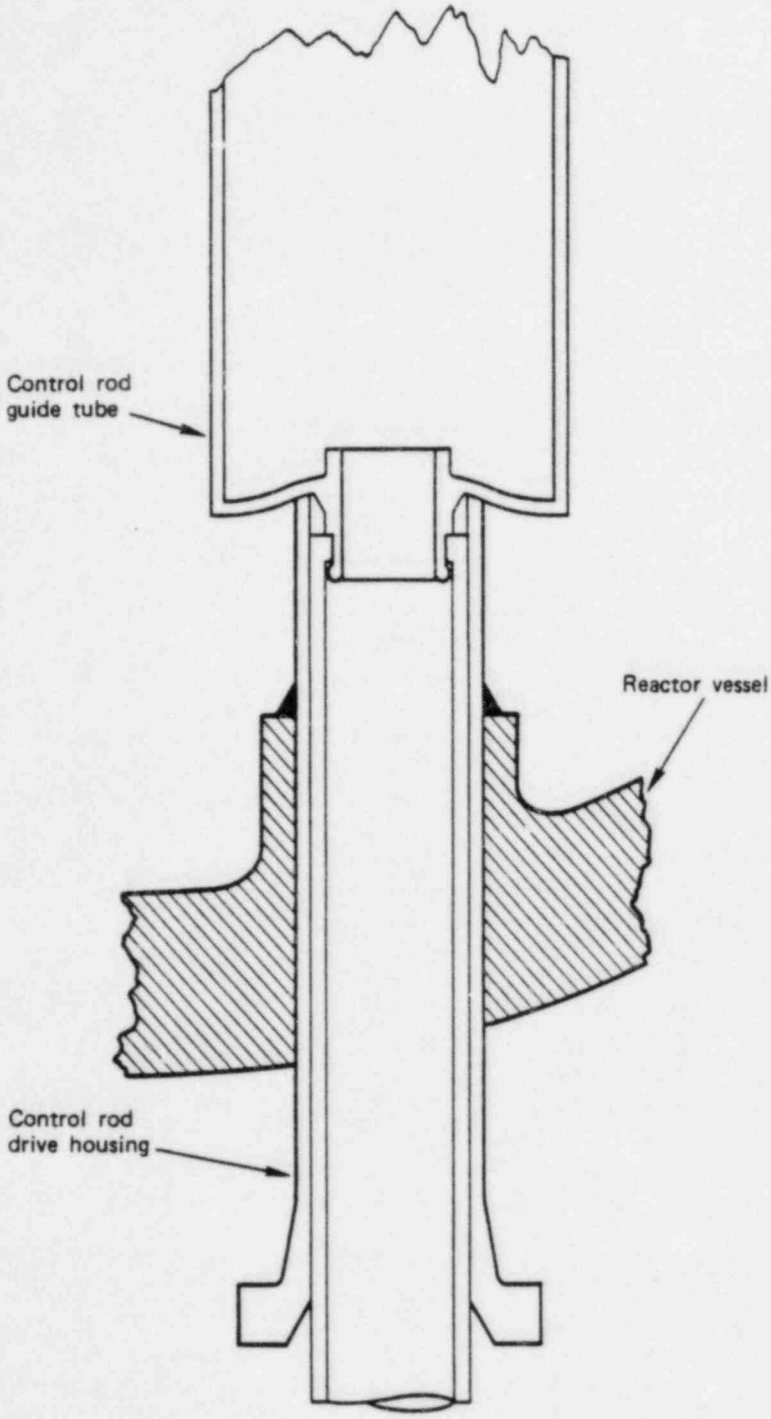


Figure 3-4. Control rod drive housing and reactor vessel penetration. Source: Spencer 1982

designed to function even in the event of complete rupture of one of the steam lines. The four systems are listed below:

- High-pressure core spray (HPCS) system (BWR/5), or high-pressure core injection (HPCI) system (BWR/4).
- Automatic depressurization system (ADS).
- Low-pressure core spray (LPCS) system.
- Low-pressure coolant injection (LPCI) system.

The HPCS system is able to use steam power to force coolant into the reactor at full system pressure while the LPCS and LPCI systems can perform a similar function at low pressure. Water can be drawn from the suppression pool, the condensate storage tank, or the borated water storage tank. The ADS will quickly reduce the primary system pressure, if activated, to a level consistent with the LPCS and LPCI capabilities. The ADS is used if the HPCS system is unable, for whatever reason, to maintain normal vessel pressure. Motor-driven pumps power both the LPCS and LPCI systems.

A number of other Engineered Safety Features are present in the Mark II plant. A partial listing includes the primary and secondary containment structures, the containment purge, the containment combustible gas control system, the containment isolation system, the service water system (tieing to the plant's ultimate heat sink), the standby gas treatment system (SGTS), and the residual heat removal (RHR) system. A number of those systems are discussed below. Attention is focused on those systems whose operation may be relevant when core degradation has occurred. Consequently, it will be assumed that the ECCS has not performed its intended function, and will not be discussed further. Appendix A describes the manner in which all the safety systems are intended to function during abnormal events, and Appendix B discusses the safety system design criteria and DBA set. The reader is reminded that the body of the report is concerned with severe accidents that are beyond the DBA set (i.e., Class 9 accident).

3.1.2 Primary Containment and Suppression Pool

The primary containment of the Mark II plant is made up of the drywell and wetwell portions (see Figure 3-1) and is essentially a large pressure vessel. The reactor, recirculating pumps, emergency core cooling equipment, and other devices are located within the upper or drywell compartment. A removable drywell closure cap permits access to the reactor for refueling operations. The suppression pool is located below in the wetwell. Steam released during an

accident to the drywell is conveyed into the suppression pool by multiple vertical steel downcomer pipes. The downcomers penetrate the diaphragm floor separating the drywell and wetwell. The suppression pool functions to condense steam and reduce primary containment pressure and temperature accordingly. It is also a water reservoir that can be tapped for emergency core cooling when necessary. The water quantity in the pool is 80,000 to 140,000 ft^3.

The primary containment design pressure is 45 to 56 psig in both the wetwell and drywell depending on the specific plant of interest. Furthermore, a negative pressure of up to -5 psig can be accommodated (subatmospheric pressures may result from containment purging and subsequent steam condensation under some accident conditions). The design leak rate is 0.5 to 1.1 percent of the primary containment volume per day at the full design pressure. Leakage is to the secondary containment where it can be processed by the SGTS for removal of radioactivity prior to venting through the plant stack. The leakage rate can be measured during normal plant operations at a pressure of 115 percent of the design pressure. The horizontal diaphragm floor is designed to withstand a downward differential pressure loading up to 30 psi. Vent valves in the floor allow free flow from the top of the wetwell back into the drywell. Table 3-3 lists many of the primary containment design characteristics for a Mark II plant.

The reactor vessel is supported on a concrete pedestal extending down to the concrete basemat of the primary containment. The diaphragm floor passes through the pedestal. The internal configuration of the pedestal immediately below the reactor varies from plant to plant (see Figure 3-5). A portion of this space is taken up by the reactor control rod drive assemblies and associated piping. The floor may simply contain drains or may be penetrated by downcomer pipes. In other plants, the diaphragm floor within the pedestal is depressed in order to create a sump.

The Mark II primary containment structures for BWR plants have been built using three basic types of construction. The first consists of concrete reinforced with multilayer deformed steel bars, a flat base, a 1/4- to 3/8-inch steel liner, and a steel dome closure cap over the reactor (Susquehanna 1 and 2, Limerick 1 and 2, and Nine Mile Point 2). A second type consists of prestressed concrete with steel cap, steel liner, and a flat base (La Salle 1 and 2, Zimmer 1). The third and last is a single containment built from steel with an ellipsoidal base (WPPS-2) (Blejwas, 1982). The steel containment is surrounded by concrete

TABLE 3-3. MARK II CONTAINMENT CHARACTERISTICS (ZIMMER PLANT)

Parameters	Values
Wetwell design pressure (psig)	45
Wetwell external design pressure, differential (psid)	2
Drywell design pressure (psig)	45
Drywell internal design pressure, differential (psid)	2
Diaphragm floor differential design pressure:	
Downward (psid)	25
Upward (psid)	3
Drywell free volume (ft^3)	180,000
Wetwell pool water volume (ft^3)	104,000
Wetwell free volume (ft^3)	93,000
Drywell design temperature (^oF)	34
Wetwell design temperature (^oF)	275
Number of downcomers	88
Downcomer internal diameter (ft)	2
Pool cross-sectional area (ft^2)	4,400
Pool depth (ft)	22
Base slab thickness (ft)	6
Inside diameter of wetwell (ft)	80
Thickness of wetwell wall (ft)	4
Thickness of drywell wall (ft)	5-6
Wetwell steel liner thickness (in)	1/4
Basement steel liner thickness (in)	1/2

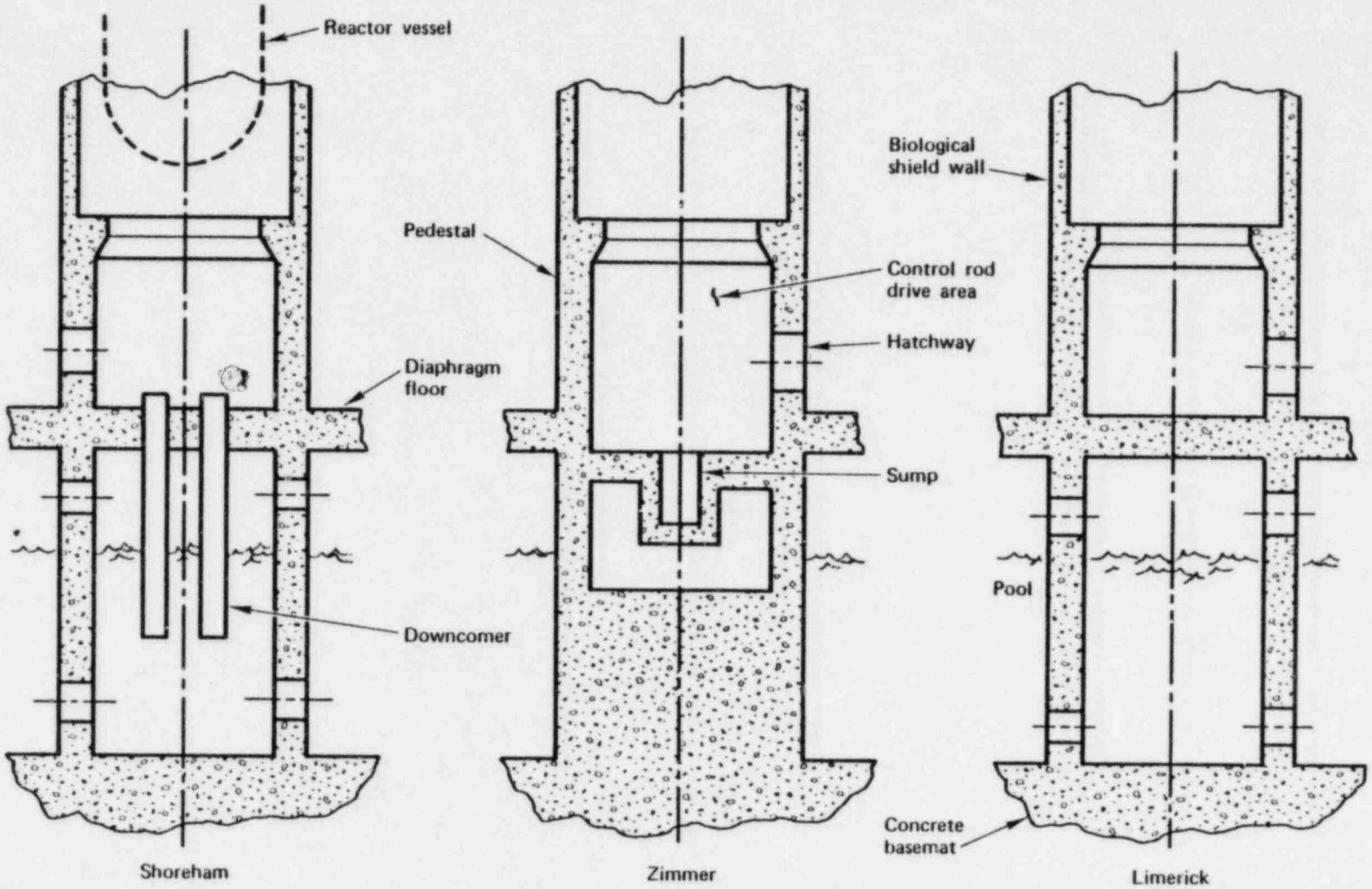


Figure 3-5. Variations in the Mark II pedestal configuration.

to create a radiation shield. Radiation shielding is inherent in the two concrete configurations.

The Mark II containments are designed to limit the leakage of radioactive materials under a combination of environmental and accident conditions. These conditions may change with time as accident loadings become better understood and new environmental loadings (e.g., seismic events) are discovered. Consequently, individual design criteria used in the various Mark II plants are not identical. The containment design process is usually based on linear elastic theory with the inclusion of safety factors at several stages in the process. Standard designs are not used, and each containment is analyzed and designed individually. The configuration of the Mark II containment is that of a truncated cone with or without a vertical cylinder segment (see Figure 3-1). A separate secondary containment (which may be the reactor building) with a low internal pressure rating surrounds the primary containment.

The advantages of the Mark II containment configuration over that of the Mark I are as follows (Wade, 1974):

1. More volume in the drywell to accommodate steam and ECCS piping as well as recirculation pumps.

2. Simpler vent configuration from the drywell to the wetwell via straight pipes.

3. Allowance for use of construction materials other than steel (like concrete).

4. Smaller reactor building.

5. Design based on an analytical model that permits a reduction in drywell and wetwell volumes.

The containments of Mark II plants are inerted with nitrogen as a form of hydrogen combustion control. This action was required as a result of the TMI-2 accident (see Federal Register, December 2, 1981). The inerting process is designed to maintain the oxygen content less than 4 percent in the primary containment. Should an accident occur that results in hydrogen production from a metal/water reaction, the resulting mixture in the containment would not be combustible or explosive.

3.1.3 Secondary Containment

The secondary containment of a Mark II plant is one of the facility's Engineered Safety Features. The reactor building

constitutes the secondary containment and, as illustrated in Figure 3-1, completely encloses the primary containment. Every Mark II plant has this secondary containment. It is designed to resist a variety of external and internal loads and, in doing so, shelter the primary containment. Internal loads result from DBA events and less severe accident conditions. External loads would result from snow, wind, flood, missiles, and earthquakes. During refueling operations, the closure cap of the primary containment is removed. Under this condition the secondary containment serves as the primary containment. It has a similar function when other maintenance operations cause the primary containment to be opened.

The secondary containment is built of steel reinforced concrete. It houses on its several floors the reactor servicing equipment, the fuel storage facilities, the ECCS equipment, the standby liquid control system, the control rod drive hardware, and the electrical equipment. Sealed hatches and other sealed penetrations isolate it from the external environment.

The internal design pressure of the secondary containment is low. Appendix E indicates that 0.25 psig is a typical value. Leakage at this pressure may be as large as 50 to 100 percent of the contained volume per day. The structure is designed so that the pressures exceeding the design value are relieved through vents located high in the secondary containment. Increasing the elevation of the release serves to maximize the opportunity for atmospheric dilution of any radioactive constituents in the vented gas.

The Mark II secondary containment is a collection point for gases leaking from the primary containment or deliberately released to it during primary containment purging. This collection action provides an opportunity to process gas for the removal of radioactivity released to the primary containment. Low level releases occur during normal operations and potentially larger ones during DBAs. The reactor enclosure recirculation system (RERS) and SGTS are automatically actuated within the secondary containment in the event of a primary containment isolation signal. Both the RERS and the SGTS are Engineered Safety Features. Two full capacity systems are provided for each. The two different systems circulate and process air within the secondary containment by filtration for the removal of halogens and particulates. Both systems are also used in the event of a refueling accident in the secondary containment. The RERS performs the initial clean-up action and the SGTS the final. The filtered exhaust from the SGTS is directed to the outside environment. In this manner it serves to keep the secondary

containment at a negative pressure during DBAs. The reduced pressure eliminates a driving force that could otherwise cause leakage through the secondary containment to the environment. The capacity of the SGTS and RERS to process secondary containment air is not large--only about 6000 scfm. Electrical power is necessary to actuate and operate both systems.

3.1.4 Isolation Devices and Protocol

The primary containment of the Mark II plant is pierced by hundreds of penetrations. These penetrations are associated with electrical power, control and instrumentation wiring; ventilation ductwork; piping; instrument tubing; and equipment hatches and personnel hatches. Sealing against leakage is provided in a number of ways and, because the primary containment is an Engineered Safety Feature, double seals or barriers are provided in most cases. The design objective for the containment isolation system is to allow normal and emergency passage of fluids into and out of the primary containment while preserving the capability of the containment to limit the escape of radioactive products released during postulated accidents. It is required that releases be constrained to meet the criteria pertaining to plant site boundary dose limits.

The containment isolation systems are actuated automatically in the event of any of the following occurrences and a corresponding signal from the plant's instrumentation network:

1. Low water level in reactor vessel.
2. High drywell pressure.
3. High temperature in the main steam line space.
4. High radiation in the steam line.
5. High flow in the main steam line.
6. Low steam line pressure at the turbine inlet.
7. High radiation levels in the reactor building ventilation exhaust.

Manual actuation is also possible from the control room and once action is initiated, the function goes to completion. The actuation signal causes all containment isolation valves to close in systems not required for emergency shutdown (the same signals activate some of the systems associated with

emergency core cooling). If necessary, the fluid lines connected to the emergency systems can be closed manually.

In general, two isolation valves are provided for all fluid lines connected to the reactor pressure boundary or containment atmosphere. One valve is located outside and one inside the primary containment barrier (small instrumentation lines that do not represent an overpressurization threat to the containment are fitted with flow-restricting orifices but not isolation valves). Valves that are pneumatically or electrically actuated are spring-loaded to fail in the closed position for nonemergency systems. The valve closure time and valve leakage rates are such that the site boundary dose criteria of 10 CFR 100 are satisfied during DBAs.

Containment isolation is also dependent on the proper functioning of numerous seals and gaskets around containment penetrations which cannot be welded to the containment itself or the containment liner. A variety of elastomeric materials are used for this sealing function. The materials, however, are susceptible to temperature, humidity, and radiation. Consequently, their performance is checked periodically when the overall containment leakage rate is measured while the containment is deliberately pressurized. The use of double seals in some instances allows individual units to be tested for leak-tightness. The containment temperature limitations are in part a consequence of the temperature sensitivity of elastomeric materials.

3.1.5 Long-Term Containment Heat Removal

Heat removal in the Mark II containment is necessary to prevent excessive temperatures and pressures during and following DBA events. The pressure must be held below approximately 50 to 70 psig, and the temperature below about 275°F, the values calculated as a result of a DBA. Heat removal may be required in either or both of the drywell and wetwell. The temperature limit is most stringent in that part of the containment containing electrical penetrations. The gaskets, seals, or packing used in these penetrations weaken rapidly at elevated temperatures. Because the heat removal system is an Engineered Safety Feature, it must operate properly even if a single failure occurs, loss of off-site electrical power takes place, adverse natural phenomena are present (e.g., earthquake or tornado), or pipes break or other accidents occur as described in the DBA set.

The containment heat removal system consists of pumps, heat exchangers, spray headers, recirculation lines, and an actuation system. It is an integral part of the RHR system.

The relevant items are arranged as shown in Figure 3-6. Water can be drawn from the suppression pool through redundant sumps, routed through one or both of the RHR heat exchangers, and discharged back into the suppression pool or to the spray headers in either or both of the drywell and wetwell. The spray headers are made of heavy steel pipe. Two mechanically and electrically independent, full capacity systems are provided. Cooling water in the RHR heat exchangers comes from the RHR service water system which ties to the plant's ultimate heat sink. The systems can be tested under normal plant operation.

The redundant RHR heat exchangers each have a capacity of about 70,000,000 Btu/hr (21 MW). Flow of suppression pool water through each unit is at a rate of about 10,000 gal/min Pumps drawing on the suppression pool may or may not require a positive suction head depending on the plant of interest.

Spray water is not directed into the drywell pedestal region beneath the reactor vessel. Rather, it is released from headers positioned high in the containment. High release improves the scrubbing effectiveness of the water. Spray water may flow into the pedestal region along the diaphragm floor in some Mark II plants. Drain holes in the diaphragm floor return water to the suppression pool for reuse.

Fan coolers provide containment temperature control during normal plant operations. These units are not Engineered Safety Features however, and may shut down automatically in the event of a DBA.

3.1.6 Combustible Gas Control

Hydrogen gas may be added to the containment following a LOCA or other events that cause accidental openings in the reactor primary system. Hydrogen is potentially available from any of the following several sources, as noted below:

1. Metal-water reactions involving the Zircaloy fuel cladding or other metals in the reactor. The Zircaloy can, under ideal conditions, produce thousands of pounds of hydrogen (see Figure 3-7).

2. Radiolytic decomposition of coolant water. This reaction also liberates oxygen.

3. Corrosion of metals and paints in the primary containment. The use of containment sprays may accelerate the corrosion process.

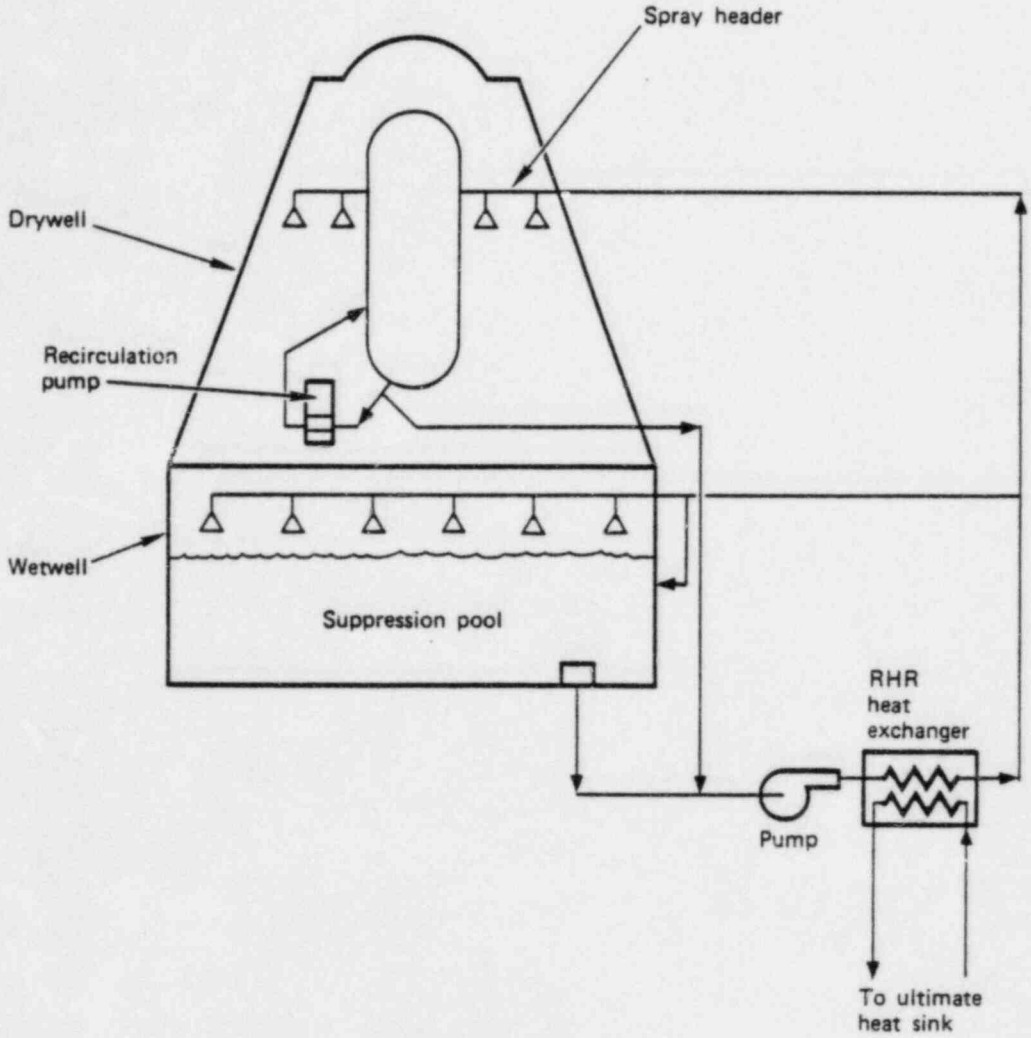


Figure 3-6. Long-term containment heat removal in Mark II plant (redundant system not shown).

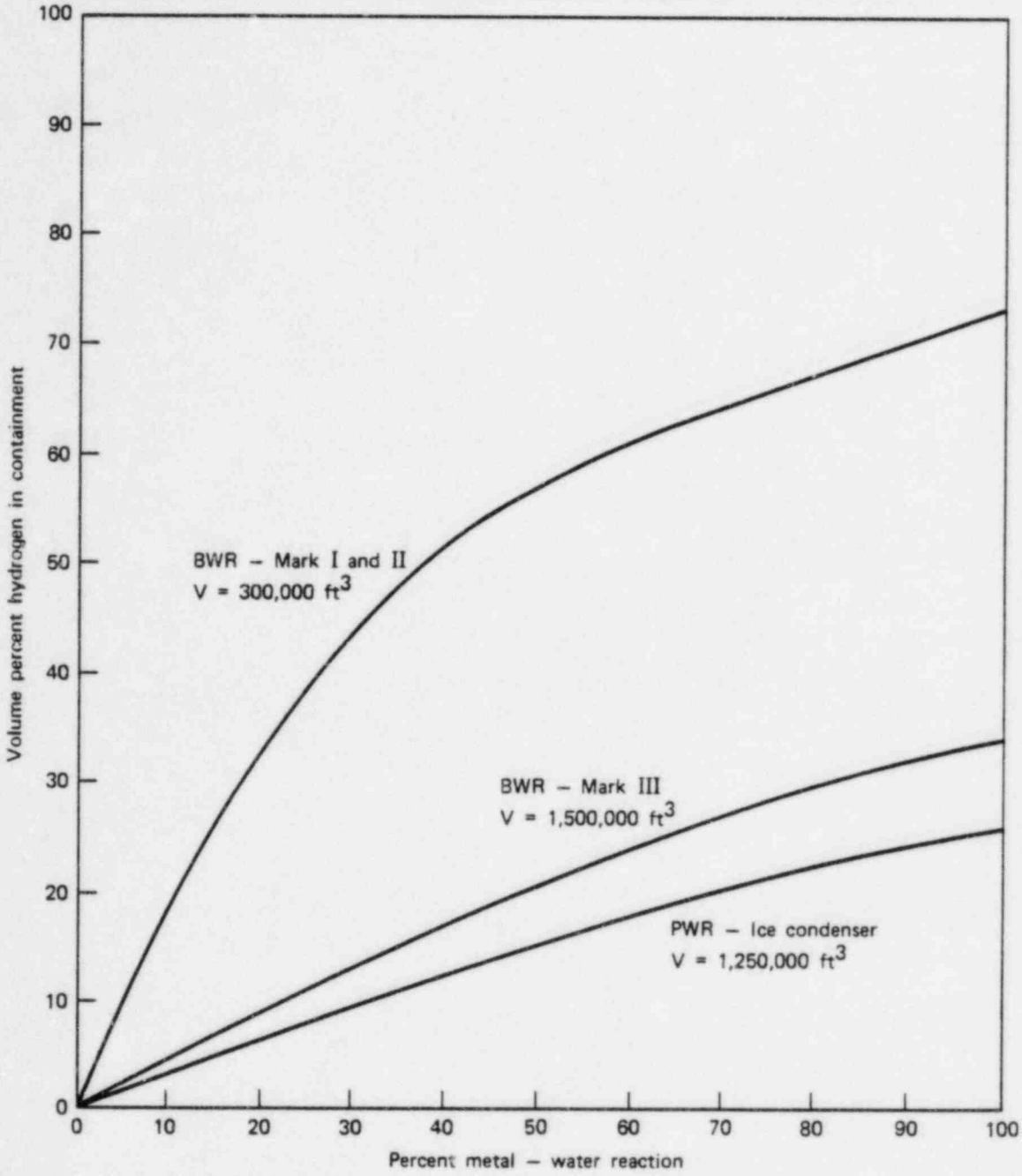


Figure 3-7. Volume percent hydrogen in containment vs percent metal-water reaction. (Source: Levy, 1981.)

The burning or detonation of hydrogen presents a serious threat to containment integrity during severe accidents (see Appendix A for a discussion of containment failure modes). Historically, 10 CFR 50 required that provisions be included in the plant to accommodate hydrogen production from the reaction of 5 percent of the Zircaloy cladding. Since the TMI-2 accident, the threat of larger hydrogen amounts has received widespread attention. As a consequence, inerting of the Mark II containment has been required.

The drywell and wetwell are inerted with nitrogen during power operation of the reactor (i.e., when the electrical output exceeds 5 percent of the rated plant capacity). The nitrogen serves to displace oxygen from the primary containment atmosphere. The concentration of oxygen is controlled to a level below the lower flammability limit. Typically, the oxygen content is held below 5 percent to satisfy this criterion. The use of inerting complicates maintenance activities within the primary containment because of the need for purging and reinerting whenever maintenance activities are conducted. It also represents a potential asphyxiation hazard to workers.

The long-term generation of hydrogen and oxygen by radiolysis is controlled by dilution and recombination. The drywell air cooling system is used to circulate air within the primary containment. This mixing action prevents the local accumulation of hydrogen and/or oxygen. An electrically operated recombiner is also present. It heats a small stream of gas drawn through it to a temperature of about 1200 F. At this temperature, the hydrogen will react with oxygen in the gas stream to form water vapor. The capacity of the recombiner is low, consistent with the gradual production of hydrogen by radiolysis.

In addition to the mixing and recombiner systems, a third aspect of the primary containment atmospheric control is a purging system. All three relate to hydrogen control and all are Engineered Safety Features. Two full capacity purge systems are available to the plant operator. They can forcefully withdraw gas from the primary containment and deliver it to the reactor building (i.e., to the secondary containment). Once in the secondary containment, the gas can be processed by the SGTS prior to its release to the environment or recycling for further hold-up in the reactor building. The SGTS acts to remove halogens and particulates from the gas. Purging is only used if the other hydrogen control measures are unsuccessful in adequately controlling hydrogen concentration in the primary containment. Additional blowers are present and can be used to return clean

outside air to the containment to make up for gas lost during purging.

3.2 DOMINANT FAILURE MODES OF THE MARK II CONTAINMENT

The Limerick PRA (Philadelphia Electric Company, 1981) and the BNL review (Papazoglou, 1982) of this PRA yield valuable insight into the dominant sequences for Mark II containments. The Accident Sequence Evaluation Program (ASEP) (Kolaczkowski, 1983) has utilized the Limerick PRA to derive a set of eight BWR accident "sequence classes" representative of dominant BWR Mark II accidents studied. Recently the Shoreham PRA (Long Island Lighting Company, 1982) has provided additional information. This section summarizes these results and presents both the dominant accident sequences and the dominant containment failure modes.

3.2.1 Dominant Accident Sequences

The dominant accident sequences leading to core melt in Mark II containments are transient events with and without scram. The dominant sequences as given in the Limerick PRA and BNL review are summarized in Table 3-4 as a function of containment class. A key to the symbols is given in Table 3-5. Both the original PRA and the review agree that the dominant initiators are transients due to loss of off-site power (T_E) and transients due to loss of feedwater or closure of the main steam isolation valve (T_F). The sequences involve loss of feedwater (Q), loss of high-pressure injection (U), failure of timely automatic depressurization (X), and failure of low-pressure injection (V), in various combinations.

The frequencies given in Table 3-4 assume that the "ATWS-3A-Fix" has been made at the plant. This fix is intended to lower the frequency of transient events without scram. From Table 3-4 it appears that these events (characterized by TC as the first two letters) represent about 3 percent of the core melt frequency. However, as will be discussed later, they may lead to early containment failure and high consequences, thus contributing significantly to risk.

In the PRAs for Mark II containments, it is convenient to bin these sequences into four containment accident classes. These are described in Section 3.2.2 below. From a frequency viewpoint, the Class I containment failure phenomena are dominant. Their relative importance from a risk viewpoint is discussed in Section 3.3.3. In the Shoreham PRA, the same four containment failure classes are employed, as well as a fifth class for LOCAs occurring outside containment. This fifth class is not dominant (frequency on the order of

TABLE 3-4. DOMINANT SEQUENCES BY CONTAINMENT FAILURE CLASS (BNL-REVIEW)

Class I		Class II		Class III		Class IV	
1. T_FQUX	3.7×10^{-5}	1. T_FQW	1.3×10^{-6}	1. T_TC_MPU	8.7×10^{-7}	1. T_TC_MD	1.4×10^{-7}
2. T_EUV	3.2×10^{-5}	2. T_TPW	7.7×10^{-7}	2. $T_FC_MU_{UR}$	5.3×10^{-7}	2. T_FC_MUD	4.0×10^{-8}
3. T_EUX	8.6×10^{-6}	3. T_EW	6.4×10^{-7}	3. $T_EC_MW_{12}$	4.3×10^{-7}	3. T_FC_MD	3.3×10^{-8}
4. T_TQUX	8.0×10^{-6}	4. T_T(WSW)	5.9×10^{-7}	4. T_IC_MU	2.9×10^{-7}	4. $T_IC_MPW_2$	2.0×10^{-8}
5. T_IUX	4.0×10^{-6}	5. T_IW	4.3×10^{-7}	5. $T_IC_MC_{12}$	2.6×10^{-7}	5. T_TC_ER	1.5×10^{-8}
6. T_T(DC)	2.0×10^{-6}	6. T_FPW	1.2×10^{-7}	6. T_FC_MPU	2.4×10^{-7}	6. $T_TC_MU_H$	1.4×10^{-8}
7. T_FQUV	1.1×10^{-6}	7. T_F(WSW)	1.1×10^{-7}	7. $T_FC_MW_{12}$	1.6×10^{-7}	7. T_TC_MPD	1.3×10^{-8}
8. T_T(AC)	6.1×10^{-7}	8. T_TQW	9.4×10^{-8}	8. $T_TC_MC_2$	1.6×10^{-7}	8. T_TC_MM	7.5×10^{-8}
9. T_T(WSW)	6.1×10^{-7}			9. $T_EC_MUU_R$	1.1×10^{-7}	9. T_TC_MR	7.4×10^{-9}
10. $T_IC'UX$	5.0×10^{-7}			10. $T_TC_MPW_2$	6.6×10^{-8}	10. T_IC_MD	3.6×10^{-9}
11. T_IUV	3.6×10^{-7}			11. $T_FC_MC_2$	4.3×10^{-8}	11. T_FC_MPD	3.6×10^{-9}
12. T_MQUX	3.6×10^{-7}			12. T_IC_MPU	3.2×10^{-8}	12. $T_FC_MU_H$	3.3×10^{-9}
13. T_F(DC)	3.1×10^{-7}			13. T_EC_MPU	2.4×10^{-8}	13. T_EC_MPD	3.0×10^{-9}
14. T_F(AC)	9.2×10^{-8}			14. $T_EC_MC_2$	2.1×10^{-8}	14. T_EC_MUD	2.7×10^{-9}
						15. T_EC_MD	2.7×10^{-9}
						16. T_FC_MM	2.1×10^{-9}
						17. $T_TC_MPU_M$	1.3×10^{-9}
Total 9.5×10^{-5}/yr		Total 4.1×10^{-6}/yr		Total 3.4×10^{-6}/yr		Total 3.0×10^{-7}/yr	

Source: Papazoglou et al., 1983.

TABLE 3-5. KEY TO LIMERICK SEQUENCE SYMBOLS

Initiating Events	
A	- Large, >4-inch diameter, LOCA
T_T	- Turbine trip with by-pass
T_E	- Loss of off-site power
T_I	- Inadvertent open relief valve
T_M	- Manual shutdown
T_F	- MSIV closure/loss of feedwater
System Failures	
AC	- Power unavailability
DC	- DC power unavailability
D,U_H	- Inadvertent operation of ADS or vessel overfill
C	- Failure to bring reactor subcritical
C_E	- Failure of the reactor protection system (electrical)
C_M	- Failure of the reactor protection system (mechanical)
C_2	- Failure of both SLC pumps
M	- Failure of safety/relief valves to open
P	- Failure of safety/relief valves to reclose
Q	- Failure of the feedwater
U,UR	- Failure of the high-pressure injection system
X	- Untimely ADS actuation
V	- Failure of the low-pressure ECCS
W	- Failure of RHR system, RCIC steam condensing, and power conversion system (PCS)
W(P)	- Event W given that P has occurred
WSW	- Loss of normal and emergency service water
R	- Failure of recirculation pump trip system
W_2,W_{12}	- Loss of containment heat removal
C_{12}	- Loss of poison injection
C'	- Untimely scram of reactor protection system

Source: Papazoglou et al., 1983.

10^{-8} per year), and the others are comparable to Limerick except the Class IV (ATWS) events. In the Shoreham PRA, the Class IV events have a frequency of 1.4×10^{-5} per year because the ATWS 3A Fix is not used.

3.2.2 Containment Failure Classes

In this section the four containment failure classes, as described in the Limerick PRA, are discussed. The containment failure classes described below are mainly caused by pressurization of the containment. This pressurization can be the result of a steam explosion, a hydrogen burn or detonation, or the production of noncondensible gases due to core-concrete interactions. Moreover, the pressurization may lead to containment relief via enhanced leakage through seals, cracks, and penetrations. In Section 3.3, the relative importance of these failure modes is discussed within the context of risk reduction.

Class I

This containment failure class is characterized by sequences initiated by transients with scram and with loss of coolant make-up to the reactor vessel. The core is expected to melt relatively fast with the containment intact and at low pressure. The containment pressure just prior to melt-through of the reactor vessel is slightly higher than atmospheric, and the suppression pool is subcooled. The containment is expected to remain intact through a large portion of the core vaporization phase. The containment is expected to fail either by overpressure in the wetwell or the drywell, or by small or large leaks. The core power at dryout is less than 2 percent full power, and the containment pressure is at 17 psi. The containment is intact at core melt and vessel melt-through. The suppression pool is still subcooled. The core melts about 1.3 hr after scram, and the vessel fails at about 4.3 hr. The diaphragm floor is penetrated about 6 to 6.5 hr following scram. Both the floor and the containment fail at this time due to overpressurization.

If significant quantities of core debris pass through the diaphragm floor shortly after vessel failure, the core would enter the suppression pool. The subsequent debris/water interactions could significantly change the time, place, and mode of containment failure. A strong interaction could result in early failure and potentially higher risk. If cooling occurs and containment failure occurs on a very long time scale, the risk would be significantly lower.

Class II

The core is expected to melt relatively slowly, and the containment is expected to be failed at the time of core melt-through. The suppression pool is assumed to be saturated at core melt. The gap release and the melt release occur through the safety relief valves to the suppression pool. The vaporization release occurs in the drywell with an open containment. The containment failure is by overpressure in the drywell or wetwell, or by small or large leaks.

This containment failure class is characterized by sequences initiated by transients with loss of containment heat removal (W). If decay heat cannot be removed from the containment building (via the suppression pool in the RHR phase), the pool becomes saturated and the containment is predicted to fail after about 30 hr due to pressurization. At containment failure, core injection is assumed to fail and the core melts. The melt attacks the vessel and eventually fails the diaphragm floor about 43.3 hr after scram.

Class III

The core is expected to melt relatively fast and the containment to fail shortly thereafter. This case is very similar to the Class I sequence except that the suppression pool is saturated during the gap and melt radionuclide releases. Therefore, the decontamination is lower for this class than Class I. This containment failure class is characterized by transients with failure to scram and with loss of coolant injection prior to containment failure. The core is assumed to be at 30 percent full power. No operator action is assumed in these ATWS events. The suppression pool is saturated prior to core melt due to steaming from the core through the safety relief valves. The containment is at high pressure (65 psi) at this stage. The core melts about 0.9 hr after initiation of the event; the vessel fails approximately 4.3 hr after initiation. The containment is calculated to fail by pressurization between 6 and 6.5 hr after initiation, depending upon the spreading of the debris on the diaphragm floor.

Class IV

The final containment failure class is characterized by transients with failure to scram, and with successful coolant injection but without adequate containment heat removal. The core is expected to melt relatively fast, with containment failure prior to core melt. This case is similar to Class II, except that the reactor is at a significantly higher power level at the time of core melt. The

containment fails by overpressure in the drywell or the wetwell. Containment failure occurs about 40 min after accident initiation. At containment failure, coolant injection is assumed to fail and the core melts about 1.2 hr after initiation. The vessel fails about 4 hr later so that the debris enters a failed containment.

Bypass Leakage from Reactor Containment

Another class of failure is potentially important, though it was not given weight in the PRAs that are the basis for this work. The reactor containment has a number of penetrations. Personnel must be able to pass in and out through an airlock and ventilation air must be supplied into and evacuated out of the containment. The closure cap must be removed and reinstalled during refueling. Feedwater enters the reactor and steam leaves it in pipes, cables for electrical power and signals must be brought out and CRD hydraulic fluid must reach the rods.

In the event of an accident with liberation of radioactive substances, passages through the containment wall are sealed by valves in pipelines. Each pipeline (or most) contains an inner and outer isolation valve for this purpose. No valve exists that does not leak. Further, normal imperfections and wear in these valves leads to slowly increasing leak rate with time. This leakage is reduced to below values specified by plant technical specifications through periodic inspection, testing and maintenance.

Conditions leading to possible high leakage rates of radioactive substances from the primary to secondary containment will differ greatly between PWRs and BWRs because the designs are different and within a given generic design type because the management of the plant differs. It is important that these individual characteristics of a plant be considered when the possibility of bypass of radioactive substances is assessed.

3.3 BENEFITS OF MITIGATION SYSTEM INSTALLATIONS

To determine the benefits derived (in terms of risk averted) from reducing the various failure modes for Mark II BWR containments, the study team used the BNL review and the Limerick PRA as a basis. It is recognized that this approach has several limitations: only internal initiators are considered, the consequences are based on RSS (NRC, 1975) source terms and methodology, there is a large uncertainty with respect to post-vessel failure phenomena, and the consequence models contain evacuation schemes

which may not apply at all sites. Moreover, the Limerick plant may not be representative of other Mark II containments, as shown in Figure 3-5.

At the end of this section, the effect of fires and seismic events as external initiators is discussed, along with the effect of the ATWS-3A-Fix. It is anticipated that during the course of this project, the consequences will be modified (based on new source term information), and new considerations of containment failure behavior will be made (time, place and mode).

The determination of risk averted for Limerick is carried out in several steps. First, the various containment failure modes are identified. These are defined in Table 3-6 for Limerick. Next, the conditional probabilities of each failure mode are determined for each core-melt accident sequence. In the Limerick PRA, the core-melt sequences are binned into four containment failure classes (depending on the relative timing of core melt and containment failure, as described in Section 3.2). The frequencies of the four classes, as well as the conditional probability of containment failure, are given in Table 3-7. Note also that the release category (the physical conditions of the radioactive material from which the consequences are calculated) is also given (in parentheses) in Table 3-7. The consequences (for acute fatalities, latent fatalities, and man-rem) for these release categories are given in Table 3-8.

TABLE 3-6. KEY TO CONTAINMENT FAILURE MODES (LIMERICK)

Symbol	Definition
α	In-vessel steam explosion
β	Ex-vessel steam explosion
γ	Overpressure: drywell failure
γ'	Overpressure: wetwell failure
γ''	Overpressure: wetwell failure, with loss of suppression pool
δ	Small leak
δ_+	Small leak, SGTS failure*
δ_ζ	Large leak
μ	Overpressure: hydrogen burn
μ'	Overpressure: hydrogen detonation

*SGTS = Standby gas treatment system.

TABLE 3-7. CONDITIONAL PROBABILITY OF CONTAINMENT FAILURE, RELEASE CATEGORY AND CLASS FREQUENCY

Mode of Containment Failure	Class I (9.5×10^{-5} yr^{-1})	Class II (4.1×10^{-6} yr^{-1})	Class III (3.4×10^{-6} yr^{-1})	Class IV (3.0×10^{-7} yr^{-1})
α	0.001 (OXRE)	0.005 (OXRE)	0.001 (OXRE)	0.01 (OXRE)
β,μ'	0.002 (OXRE)	0.05 (OXRE)	0.002 (OXRE)	0.09898 (OXRE)
γ,μ	0.2560 (OPREL)	0.2245 (OPREL)	0.256 (OPREL)	0.445 (C4γ)
γ',μ	0.1235 (OPREL)	0.1105 (OPREL)	0.1235 (OPREL)	0.2226 (C4γ')
γ'',μ	0.1235 (OPREL)	0.1103 (OPREL)	0.1235 (OPREL)	0.2226 (C4γ'')
δ	0.2223 (none)	0.500 (none)	0.2223 (none)	-- --
δ_{τ}	0.0247 (OPREL)	-- --	0.0247 (OPREL)	-- --
δ_{ξ}	0.247 (OPREL)	-- --	0.247 (OPREL)	-- --
Total	1.000	1.000	1.000	1.000

Definitions:

OXRE is the oxidation release.

OPREL is the overpressurization release.

C4γ is failure of the drywell release for ATWS.

C4γ' is failure of the wetwell above the suppression pool release for ATWS.

C4γ'' is failure of the wetwell below the suppression pool release for ATWS.

TABLE 3-8. CONSEQUENCES FOR EACH RELEASE CATEGORY (BNL REVIEW)

Release Category	Acute* Fatalities	Latent* Fatalities	Man-Rem* (500 miles)	Man-Rem* (50 miles)
OPREL	0	2.2×10^3	1.42×10^7	0.78×10^7
OXRE	97	1.9×10^4	4.90×10^7	2.5×10^7
C4γ	75	1.4×10^4	7.88×10^7	4.7×10^7
C4γ'	69	1.4×10^4	7.86×10^7	5.3×10^7
C4γ"	138	1.3×10^4	7.36×10^7	3.6×10^7

*Based on WASH-1400 Source Terms Data and Methodology.

The last step is to determine the potential risk that can be averted if each containment failure mode is eliminated. In some sense, this assumes perfect mitigation and hence gives an upper bound on potential benefit. For the Limerick Plant, it is convenient to use Eq. (3) given in Section 2.2 of this report. The sequence frequencies for each containment failure class have been summed and are given in Table 3-4. These are multiplied by the conditional probability of containment failure given in Table 3-7 and the corresponding consequence as given in Table 3-8. This operation can be expressed as:

$$\text{Risk Averted} = f_j \times P_{jk} \times C_{ki}$$

where j denotes the jth sequence, k denotes the kth containment failure mode, and i denotes the ith consequence of interest. The symbols f, P and C are, respectively, the containment accident class frequency, the containment failure mode conditional probability, and the consequence.

Tables 3-9 to 3-11 contain the values of risk averted for the following consequences: man-rem (Table 3-9), acute fatalities (Table 3-10), and latent fatalities (Table 3-11), assuming that the "ATWS 3A Fix" has been committed to by the plant owner. A calculation that omits the ATWS-3A-Fix will be considered later. These three tables can be used to define the potential for accident mitigation at Limerick.

TABLE 3-9. MAN-REM/YEAR AVERTED FOR EACH CONTAINMENT FAILURE MODE - INTERNAL INITIATORS, ASSUMING PERFECT MITIGATION (WITH ATWS-3A-FIX)*

Failure Mode	Class I	Class II	Class III	Class IV
α	2.6	0.56	0.09	0.08
β,μ'	5.2	5.6	0.18	0.08
γ,μ	193.2	7.3	6.9	6.3
γ',μ	93.0	3.6	3.3	3.5
γ'',μ	93.0	3.6	3.3	2.4
δ	--	--	--	--
δ_τ	18.5	--	0.67	--
δ_ξ	186.5	--	6.7	--
Total	592.0	20.7	21.1	12.4
Total potential risk averted = 646 man-rem/year to 50 miles				

*Based on WASH-1400 Source Terms and Methodology.

Examination of Table 3-10 shows that reduction of acute fatalities requires mitigation of the steam explosion for Class I and Class II sequences, and the elimination of overpressure failure for ATWS-Class IV events. It should be noted, however, that the OPREL release assumes that evacuation occurs prior to late overpressure failure leading to no acute fatalities. This may be nonconservative because evacuation may be delayed in some dominant Class I sequences where off-site power is lost. Examination of Table 3-11 shows that the reduction of latent fatalities is best achieved by eliminating the slow overpressurization failure of containment for Class I sequences. Table 3-9, which displays man-rem averted, confirms this observation.

Table 3-12 shows the dollar value (based on $1000 per man-rem averted) for perfect (100 percent effective) reduction of risk for each accident class and each containment failure mode. Examination of the table shows that the biggest potential is elimination of the slow overpressurization failure modes for Class I sequences ($15.1 million). The second highest in value is the elimination of large containment leakage for Class I sequences ($13.3 million).

TABLE 3-10. ACUTE (EARLY FATALITIES)/YEAR AVERTED FOR EACH CONTAINMENT FAILURE MODE - INTERNAL INITIATORS (WITH ATWS-3A-FIX) AND ASSUMING PERFECT MITIGATION*

Failure Mode	Class I	Class II	Class III	Class IV
α	0.9×10^{-5}	2.0×10^{-6}	3.3×10^{-7}	2.9×10^{-7}
β,μ'	1.8×10^{-5}	2.0×10^{-5}	6.6×10^{-7}	2.9×10^{-6}
γ,μ	0	0	0	1.0×10^{-6}
γ',μ	0	0	0	4.6×10^{-6}
γ'',μ	0	0	0	9.3×10^{-6}
δ	0	0	0	0
δ_{+}	0	0	0	0
δ_{ξ}	0	0	0	0
Total	2.7×10^{-5}	2.2×10^{-5}	0.1×10^{-5}	2.7×10^{-5}
Total potential risk averted = 7.7×10^{-5} acute fatalities/year				

*Based on WASH-1400 Source Terms and Methodology.

TABLE 3-11. LATENT FATALITIES/YEAR AVERTED FOR EACH CONTAINMENT FAILURE MODE - INTERNAL INITIATORS (WITH ATWS-3A-FIX) AND ASSUMING PERFECT MITIGATION*

Failure Mode	Class I	Class II	Class III	Class IV
α	0.18×10^{-2}	0.39×10^{-3}	0.65×10^{-4}	0.57×10^{-4}
β,μ'	0.36×10^{-2}	0.39×10^{-2}	1.3×10^{-4}	0.56×10^{-3}
γ,μ	5.3×10^{-2}	0.20×10^{-2}	1.9×10^{-3}	1.8×10^{-3}
γ',μ	2.6×10^{-2}	0.99×10^{-3}	0.92×10^{-3}	0.94×10^{-3}
γ'',μ	2.6×10^{-2}	0.99×10^{-3}	0.92×10^{-3}	0.88×10^{-3}
δ	0	0	0	0
δ_{+}	0.5×10^{-2}	0	1.85×10^{-4}	0
δ_{ξ}	5.2×10^{-2}	0	1.84×10^{-3}	0
Total	16.8×10^{-2}			
Total potential risk averted = 18.6×10^{-2} latent fatalities/year				

*Based on WASH-1400 Source Terms and Methodology.

TABLE 3-12. VALUE OF MITIGATION BASED ON $1000/MAN-REM AVERTED (OUT TO 50 MILES) (INTERNAL INITIATORS ONLY) IN MILLION DOLLARS (BASED ON 40-YEAR PLANT LIFE) AND NO DISCOUNTING

Failure Mode	Class I	Class II	Class III	Class IV	Total
β	0.09	0.020	0.003	0.003	0.116
β,γ	0.19	0.208	0.007	0.003	0.408
γ,μ	7.73	0.291	0.274	0.252	8.55
γ',μ	3.7	0.146	0.134	0.141	4.12
γ'',μ	3.7	0.146	0.134	0.100	4.08
δ	--	--	--	--	--
δ_+	0.73	--	0.027	--	0.76
δ_ξ	7.45	--	0.266	--	7.72

3.3.1 ATWS-3A-Fix

The ATWS-3A-Fix is intended to reduce the frequency of containment Class III and IV sequences. The BNL review gives the frequency for Class III and IV sequences with and without the ATWS-3A-Fix respectively. These frequencies were used to construct Table 3-13, which shows the risk in man-rem for each containment failure mode by accident class. As expected, the most notable changes are the additional contribution to the slow-overpressurization modes for the Class III sequences and the large leaks. The value of mitigation for these modes (at a $1000 man-rem averted out to 50 mi and assuming a 40-year plant life with no discounting) is $5.7 million as compared to $0.5 million with the ATWS-3A-Fix; the fix itself is worth $9.0 million. The presence of the ATWS-3A-Fix does not affect the potential for mitigating slow overpressurization. Without the fix, the value of mitigation in terms of a risk reduction factor would be enhanced.

3.3.2 External Events

The Severe Accident Risk Assessment (SARA) for the Limerick Plant (Philadelphia Electric Company, 1983) shows that the major contributors to risk from external events are fires and earthquakes. Fires increase the frequency of Class I

TABLE 3-13. MAN-REM/YEAR AVERTED FOR EACH CONTAINMENT FAILURE MODE - INTERNAL INITIATORS (WITHOUT ATWS-3A-FIX) ASSUMING PERFECT MITIGATION*

Failure Mode	Class I	Class II	Class III	Class IV
α	2.6	0.56	0.94	0.20
β,μ'	5.2	5.6	1.8	0.20
γ,μ	193.2	7.3	72.8	16.8
γ',μ	93.0	3.6	35.3	9.52
γ'',μ	93.0	3.6	35.3	6.5
δ	--	--	--	--
δ_{+}	18.5	--	7.28	--
δ_{ξ}	186.5	--	72.2	--
Total	592.0	20.7	225.62	33.22
Total potential risk averted = 872 man-rem/yr, to 50 miles.				

*Based on WASH-1400 Source Terms and Data.

sequences by 24 percent, while seismic initiators increase the frequency of Class I and III sequences by 3 percent and 26 percent, respectively. Class II and IV sequences are also affected, but as discussed above, the potential for mitigation is in the Class II and III sequences. Seismic events can also fail containment; however, these events are considered uncontrollable and are therefore treated separately. The inclusion of fires and seismic initiators provides additional opportunity for mitigation as shown in Table 3-14.

The inclusion of external events for Limerick increases the value of perfect mitigation by about 25 percent. Achieving this extra 25 percent requires that the mitigation system be designed to withstand the threat of these external events, however. Note that for this chapter, risk reduction factors were not calculated. In Chapter 8, risk reduction factors are calculated and compared to other plants.

TABLE 3-14. VALUE OF MITIGATION FOR LIMERICK ASSUMING 40-YEAR PLANT LIFE AT $1000/MAN-REM AVERTED AND PERFECT MITIGATION

OVERPRESSURIZATION				
WITH ATWS-3A-FIX IN MILLIONS OF DOLLARS				
	Internal	Fires	Seismic	Total
Class I	$15.13	$3.6	$0.45	$19.2
Class III	0.54	--	.13	0.67
	$15.7	$3.6	$0.58	$19.9
WITHOUT ATWS-3A-FIX IN MILLIONS OF DOLLARS				
	Internal	Fires	Seismic	Total
Class I	$15.13	$3.6	$0.45	$19.2
Class III	5.7	--	1.5	7.2
	$21	$3.6	$2.0	$26.4

3.4 MITIGATION OPPORTUNITIES FOR MARK II CONTAINMENTS

The previous section showed that significant public risk can be averted if mitigation of Class I and Class III containment failures could be accomplished, and that more than $20 million might in principle be justified to this end. Mitigation of Class II and IV containment failures provide little overall risk reduction. To better characterize the opportunities thus far presented, the cumulative principle described in Chapter 2 is applied in order to assemble the envelope of conditions under which mitigation might have to operate. Following a description of the envelope of conditions, a set of ground rules is presented that have been adopted for handling uncertainties in phenomenological information and other conditions of the study. The specific accident end-states that are expected to result in containment failure are listed next, and from these are derived the cumulative list of functions that a mitigation system must perform. Finally, various alternatives for performing the required functions are discussed.

3.4.1 Cumulative List of Accident Conditions in Mark II Containments

During a severe accident of the Class I or III type, any or all of the following conditions are assumed to exist in conjunction with a core-melt accident.

1. All electric power for operating the prevention and mitigation equipment has been lost. Controls and instruments may be inoperative.

2. The suppression pool water has been saturated at containment design pressure and temperature.

3. The normal and emergency core cooling systems are not functioning.

4. The normal containment heat removal system is inoperative.

5. The core is essentially dry; the temperature has increased to the point at which collapse of the core has begun; and melt-through of the reactor pressure vessel (RPV) wall will occur shortly.

6. Molten steel will accompany the core debris to the retention area--perhaps as much as 50 tons.

7. Most of the Zircaloy cladding of the fuel rods will have reacted with hot steam to form hydrogen gas--estimated up to 95 percent.

8. Plant operators are unavailable to initiate any mitigative action. (Note: This is a correct assumption based on present plants and operating rules. If specific mitigation actions can be defined and adopted, the assumption would be changed accordingly.)

9. If the ATWS scenario is to be included in the mitigation design (i.e., where the 3A-fix has not been previously installed as in the Shoreham PRA), then another condition must be added, wherein a high steam generation rate for possibly tens of minutes before control is restored or the core boils dry and is damaged. Even if control is regained, the containment may be fully charged with thermal energy. If control is lost, the core-melt accident conditions noted above result.

3.4.2 Criteria for Mitigation Choices

The philosophy adopted here in applying accident mitigation is to ensure as positive and defensible a result as possible. This objective establishes the following additional requirements or conditions:

1. Whenever the accident end-state conditions are uncertain and the technical community is not in agreement as to the outcome, the anomalous situation will be circumvented by a design that either avoids or minimizes these uncertainties. The result is a known end-state.

2. Passive mitigation systems will be used whenever possible. Where impossible or unreasonably costly, a quasi-passive system requiring no personal attention will be used. A fully independent, highly reliable and dedicated source of energy will be used to perform the task.

3. Controlled venting of the containment is always preferable to any form of uncontrolled failure of the structure or its penetrations. An intact containment presents less risk to the area than a ruptured one. Vents should automatically open as specified to eliminate a value judgment by a single individual. Such vents should use a reclosing-type valve that releases only the minimum amount of gas or vapor necessary to keep the containment pressure within prescribed limits.

4. Maintaining directional control and adequate cooling of the errant core debris mass at all times presents less risk and less future clean-up cost than allowing any of it to escape into the environment or to indefinite locations in the containment interior where its subsequent behavior is uncertain.

5. Mitigation equipment and its operation should present minimal interference to the normal operation and maintenance of the plant.

6. Mitigation equipment should be of heavy duty industrial grade quality. Where systems penetrate containment, they must satisfy the usual regulatory requirements (e.g., general design criteria).

7. Where possible, alternative means of handling these accident conditions will be presented and

evaluated. Carefully defined operator actions could be such an alternative.

3.4.3 Potential Accident End-States to be Mitigated

The following accident end-states define the initial set of possible containment threats:

1. ATWS steam generation. The end-state results in an energy charged containment system and overpressure with a high steam generation rate.

2. In-vessel hydrogen generation. The end-state results in additional noncondensibles in the containment. It also presents the possibility of a hydrogen burn or explosion when the containment is deinerted (about 10 percent of the time).

3. Containment concrete decomposition. The condition results in steam and carbon dioxide to add to the already high containment loading.

4. Ex-vessel steam pressure rise when the hot core debris encounters water. A short term but high rate of steam generation occurs when residual sensible heat in the core debris mass is released to water, resulting in possible containment overpressure.

5. Ex-vessel steam explosions. While this phenomenon is still very much in controversy among the technical community, our philosophy requires that we consider it initially as an assault to be dealt with in accident mitigation.

6. Ex-vessel hydrogen generation. This end-state results when hot steel and any remaining Zircaloy in the debris mass contact hot steam and react, adding combustible noncondensibles to the gas loading of the containment.

7. Residual heat load. This condition occurs from the radioactive fuel decay energy and can result in a containment overpressure in the long term.

3.4.4 Cumulative List of Mitigation Functions to be Performed

A complete mitigation system must be capable of the following functions. However, the actual choice depends on the results of a value/impact analysis for a specific plant.

1. Containment venting with overpressure relief valves to release relatively clean ATWS steam to the atmosphere and a diverting system to pass later and generally smaller flows of contaminated steam and gas through a condenser/filter system.

2. Adequate long-term heat removal from the containment during the accident and as long as residual heat in the core is being generated.

3. Core debris mass control during its course from the RPV to a long-term retention area.

4. Adequate cooling of the core debris once it leaves the RPV.

5. Vacuum breaker system to preclude containment underpressure as steam in the containment is condensed.

6. Hydrogen control from the onset of the core-melt accident and as long as necessary afterward--generally until the combustible contents are below a flammable range, <u>if the containment is deinerted</u>. Alternatively, reduce the amount of time operation is permitted with a deinerted containment.

7. Missile shields for protection of seals and penetrations to protect against failure due to steam explosions.

3.4.5 Candidate Means for Meeting Requirements

Mitigation devices and components known in the literature and potentially suitable for the above requirements are discussed below, with indications of cost range where possible.

Containment Heat Removal

Nearly all the accident sequences result in an overheated, overpressurized containment structure with water vapor being a large part of the total pressure. Venting to relieve this condition is only a delaying action, as the residual core heat generated will boil off all water and eventually fail the containment structure, releasing radioactivity to the environment. Heat removal is thus an essential function; three distinct alternatives have been considered:

1. Heat pipes with input surface in the drywell region and discharge surface to the atmosphere outside. Heat pipes have the distinct advantage that their operation is completely passive, using only natural thermal convection forces at both ends. Their disadvantage is that a large surface area is required, many new penetrations of the containment wall are necessary, and their cost is excessive (Ahmad et al., 1983).

2. Cold water spray condensers in the drywell. Such sprays would be a conveniently installed, effective means of condensing steam in the drywell. They would serve a valuable additional function of condensing iodine vapors, washing down the drywell walls, and reducing the spread of contamination (Ahmad et al., 1983). They are modest in cost. The water supply to feed these sprays could be taken from the suppression pool or from external cool sources. If taken from the pool, the water must be cooled since it may be at saturation temperature. The rejected heat will go to an external pond, the cooling tower pool or some other ultimate heat sink. Water supplied to the sprays from an external source is a low cost and simple alternative, but this excess water adds to the contaminated inventory that must be processed eventually, reduces the containment free volume, and cannot be continued indefinitely. All the spray options require a source of power to provide water under pressure. A completely nonelectric isolated pumping plant using direct-drive diesel engines is preferred as a power source. Such a plant, dedicated to heat removal only, can operate automatically despite complete plant electrical failure.

3. Surface-type heat exchangers to cool suppression pool water. Heat could be removed directly from the suppression pool without sprays by forced circulation of the pool water through heat exchangers. This would be an adequate system by itself provided the entire core arrived promptly in the pool and not partially in the drywell, and that other gas formation and steam spikes could be tolerated without the quenching action provided by the sprays. On the other hand, cooling the pool alone has the disadvantage that the slow transfer of heat and vapor from the upper parts of the drywell into the pool may be a limiting condition.

Containment Venting

Preventing an uncontrolled rupture of the containment is an essential part of any mitigation policy. This can be done by venting clean steam and nitrogen directly to the surroundings should an ATWS event occur, while venting smaller quantities of contaminated steam and gas later through condensers and filter beds. A number of filtered vent options of general applicability have been examined (Murfin, 1980; Levy, 1981; Ahmad et al., 1983; and Reilly, 1982) as well as designs suitable for the Mark II (Levy, 1981; Reilly, 1982). A large capacity design is described by Benjamin (1981).

Core Retention or Debris Control

Ex-vessel control of the core debris is required for almost every severe accident scenario. For a core-melt accident, the final outcome is inevitably the descent of the hot core debris into the pedestal region below the RPV or on the diaphragm floor. Proper initial quenching and cooling are essential to minimize not only hydrogen generation and concrete decomposition but also ex-vessel steam spikes and core overheating. Permanent retention on the diaphragm floor or in the pool, and cooling of this material are desirable in all long-term mitigation concepts. It should be noted that the decision on a core retention system for Mark II containments will be very plant specific because of variations in the pedestal designs (See Fig. 3-5).

Combustible Gas Control

Hydrogen control is already provided in the Mark II by preinerting the containment with nitrogen. Additional measures for hydrogen control may be needed (Papazoglou, 1982) to reduce the danger of flammability during service deinerting. These measures might include burning after the deliberate addition of oxygen to the containment.

Containment overpressure from internal combustion can be provided by one of the following methods:

1. Deliberate ignition of hydrogen gas (small burns).
2. Oxygen removal (inerting).
3. Suppression of ignition (Halon, fogs, foams).
4. Hydrogen removal (recombining).

Increased Containment Mass Holding Capability

This function could be provided by the following:

1. Increased volume.
2. Increased pressure capability.
3. Improved pressure-suppression capability.

Protection for Containment Penetrations

The containment should be provided adequate protection from internal missiles and local excessive temperatures.

3.4.6 Concluding Remarks about Candidate Mitigation Devices

An appropriately selected component from each of the above six categories would adequately mitigate the consequences of a severe accident in a Mark II containment. However, not all the categories are necessarily justified in view of their cost versus risk averted. Before specific systems can be selected, each component must be assessed for practicality ity, reliability, availability, and risk reduction effectiveness at a specific site. The performance of the components together in forming the final system must receive a similar assessment, since some components can perform more than one function.

The above discussion of mitigation options is strongly focused on special hardware to be retrofitted to the Mark-II-type plants. While it is essential to define this aspect carefully to assure that a reliable, cost-effective mitigation option exists, the possibilities do not end here. The essential mitigation functions can to some extent be performed by upgrading and modifying existing plant equipment, and by carefully training operators to take corrective action in the postaccident situation. Details of the costs of such mitigation steps can be determined only by careful on-site engineering design and by conducting PRAs of the resulting change in accident consequences. Although the facilities to accomplish these assessments are not available at present, they would represent a useful part of future mitigation studies. In general, one can say that both the cost and the effectiveness of such measures should be less than those of the dedicated hardware discussed above.

As has been mentioned previously, severe accident consequence mitigation is just one of several ways of addressing risk. One of the other approaches focuses on improving the adequacy of the plant's Engineered Safety Features. The intent in this is to augment the likelihood of successfully

preventing severe accidents. Improvements in accident prevention are thought to be possible because of deficiencies discovered in the existing Engineered Safety Features.

3.5 SUMMARY

This review of the severe accident response of the Mark II containment system shows that it is a compact, efficient design that is well adapted to its design basis. The analysis shows that some modifications would extend its capability to deal with accidents that exceed the design basis, and that reduction in the residual public risk would result therefrom.

As with most light-water reactors (LWRs), long-term slow overpressurization represents the main element of risk. However, for a Mark II system the most unusual aspect in mitigating the effects of severe accidents is the possibility that a core meltdown might emerge into a containment already at design pressure with a saturated suppression pool. One pathway to such a condition is through a sustained ATWS event; others include sequences beginning with a turbine trip or pipe rupture. Even though the proposed ATWS-3A-Fix would reduce the probability of an ATWS event, the overall residual risk appears to be sufficient to justify provision for venting the clean steam from the ATWS, as well as provision against hydrogen formation (as a noncondensible gas) and concrete attack when the suppression pool is already saturated at design pressure.

Steps for mitigating severe accidents for this containment include (1) cooling the containment, (2) preventing concrete attack, (3) venting contaminated noncondensibles from a meltdown at high containment pressure, and (4) venting clean steam from an ATWS condition. These are listed in order of decreasing benefit if considered singly, but omitting any one may substantially decrease the value of all the rest.

Preliminary assessment of the risk averted with perfect mitigation, using the $1000/man-rem averted algorithm, shows that significant amounts are indicated as the value of mitigation in this containment. If the ATWS-3A-Fix is evaluated as effective to the extent assumed above, the cost justified for its installation is about $10 million. Costs justified for mitigation range from $26.4 million without the 3A, to about $19.9 million with it. Well-developed mitigation equipment is available to meet the requirements, but the questions of cost and compatibility with existing systems remain to be explored in a more detailed study. These results are summarized in Table 3-15.

TABLE 3-15. SUMMARY OF MITIGATION OPPORTUNITIES* (VALUE OF MITIGATION AT $1000/MAN-REM, NO DISCOUNTING)

Case Numbers	Allowable Cost ($M)
1. Mitigate all Class I overpressure failures ($\gamma, \gamma', \gamma'', \mu$)	$15
2. Mitigate large containment leakage Class I (δ_{ξ})	$ 7.5
3. Mitigate small containment leakage Class I (δ_{+})	$ 0.7
4. Mitigate all OPREL releases	$25.0
5. Mitigate Class III overpressure failures ($\gamma, \gamma', \gamma'', \mu$)	$ 0.54
6. Mitigate Class I and III	$24.5
7. Value of ATWS-3A-Fix	$10.0
8. Mitigate Class III overpressure failure ($\gamma, \gamma', \gamma'', \mu$) without 3A-Fix	$ 5.7
9. Mitigate leakage without ATWS-3A-Fix	$10.3
10. Include external events with ATWS-3A-Fix	$19.9
11. Include external events without ATWS-Fix	$26.4

*Based on the Limerick PRA, RSS source-term methodology.

CHAPTER 4. THE BWR MARK III CONTAINMENT

The Mark III containment for the GE BWR plants is the most advanced BWR-type containment to be built to date. As shown in Appendix D, twelve plants use this type of containment category. However, only one plant--Grand Gulf 1 in Port Gibson, Mississippi--has been granted an operating license at this time. The other units are either still in construction or in the planning stages. All have large electrical generating capacities, ranging from 933 to 1233 MWe. As with all other reactors, the Mark III containment is an Engineered Safety Feature and, as such, is intended to withstand DBAs.

Figure 4-1 is a section view of a typical Mark III containment. The major features are the primary containment itself, the secondary containment (also known as the reactor enclosure or shield building), the reactor vessel, the interior drywell with overhead sprays, and the suppression pool. The primary containment may be constructed of either heavy steel plate or reinforced concrete with a continuous steel membrane liner. The drywell serves to channel steam from an accidental primary reactor system blowdown into the suppression pool through one or more submerged vents in the lower drywell wall.

The Mark III retains the pressure-suppression function embodied in the Mark I and II plants of earlier design. However, as illustrated in Figure 4-1, the new drywell wall is not part of the primary containment barrier. The cylindrical drywell wall supports the upper rectangular refueling pool and intermediate floors within the reactor building. The containment is similar in size and design to a typical dry containment, whereas the Mark I and II configurations are considerably smaller and of different shapes (see Appendix E). A secondary containment is usually provided as an integral part of the Mark III plants in order to restrict radiation releases in the event of a DBA. The reactor building encloses the following elements of the plant: (1) shield structure (i.e., the concrete immediately adjacent to the primary containment wall), (2) the primary containment, (3) the reactor, (4) the drywell, and (5) the suppression pool. The refueling building and auxiliary building (containing ECCS and electrical equipment) are separate structures and are not designed as Engineered Safety Features.

The pressure-suppression feature of the Mark III plant, though arranged in a physically different manner, retains the functions of fission-product scrubbing, steam condensation, and heat sink as in the Mark I and II plants.

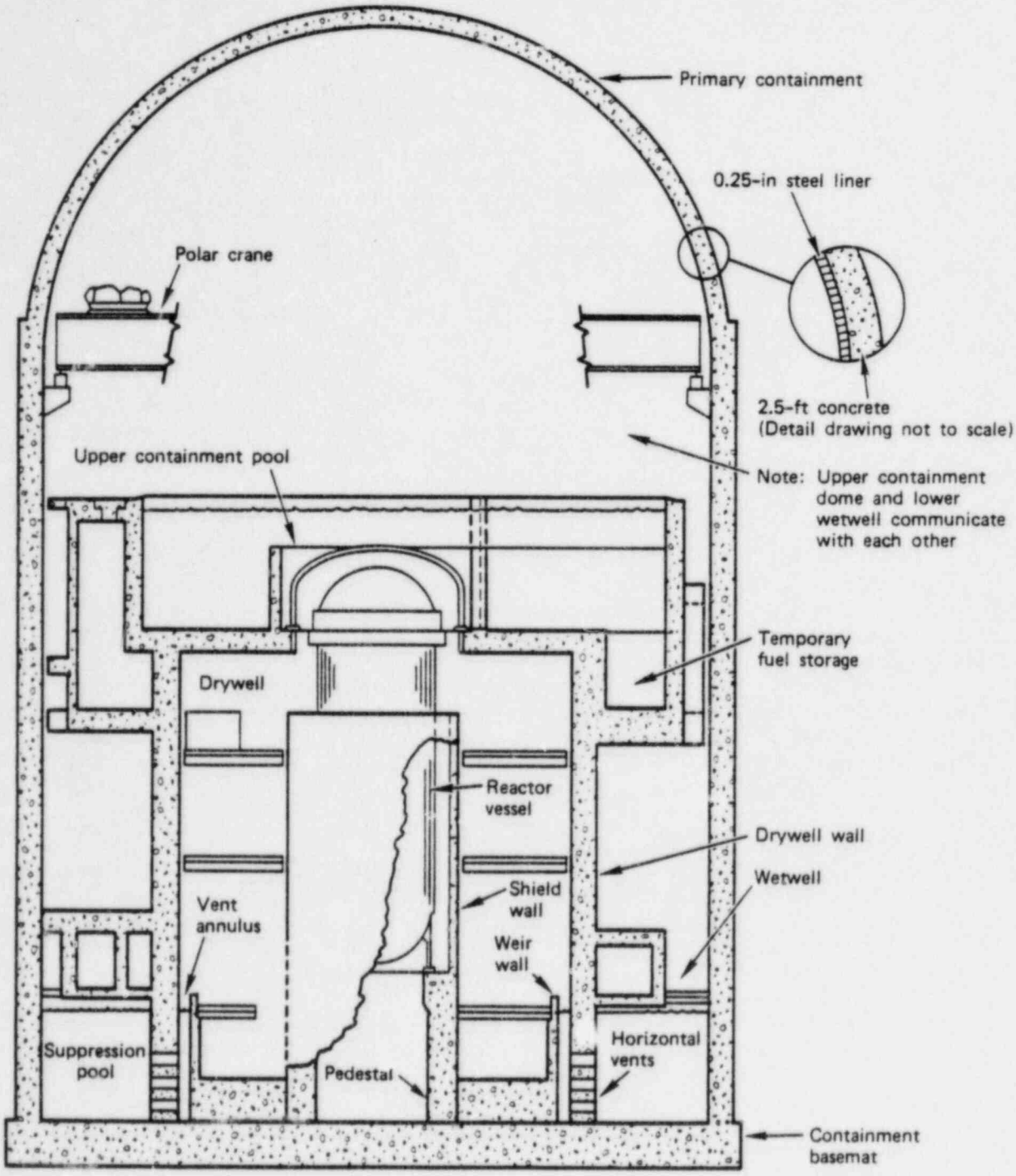


Figure 4-1. Mark III containment.

Table 4-1 compares some of the major containment characteristics of the Mark III with the Mark I and II plants. Because of the large volume of the Mark III containment, it has not been necessary to inert the containment to maintain hydrogen concentrations below the explosive limit in the event of a metal/water reaction involving up to 5 percent of the fuel cladding (as required by Regulatory Guide 1.7).

The Mark III containment is described in further detail in Section 4.1, along with the other Engineered Safety Features that complement its role in controlling the release of radioactivity during serious accidents. Reference is made to the assumed DBAs that are crucial to the containment design; more severe accidents that can challenge containment integrity are also identified. The important accident sequences pertaining to the potential loss of containment integrity are drawn from PRAs prepared for the Mark III plants (Section 4.2). Specific reference is made to the Grand Gulf PRA in this exercise.

Mitigation systems are proposed in Section 4.4 that may be able to alleviate the consequences of the Class 9 accidents. The extent to which this is theoretically possible is estimated along with the value of such a risk reduction. The calculation of changes in risk is made through modifications of the PRA results. It is assumed in Section 4.3 that certain release categories can be wholly eliminated through the action of selected ideal mitigation systems (later program tasks will estimate the actual extent to which real-world mitigation systems can, in fact, match this ideal

TABLE 4-1. BWR CONTAINMENT CHARACTERISTICS

Parameters	Mark I (Peach Bottom)	Mark II (LaSalle)	Mark III (Hartsville)
Drywell free volume (ft^3)	176,000	221,000	274,000
Suppression pool free volume (ft^3)	128,000	166,000	1,140,000
Water volume (ft^3)	123,000	143,000	130,000
Drywell deisgn pressure (psig)	62	45	30
Wetwell design pressure (psig)	62	45	15

performance). The value of risk reduction is computed on the basis of the change in man-rem caused by eliminating release categories. Based on the value of various mitigation systems, recommendations are made for further detailed study of mitigation systems in future program tasks (Section 4.5).

4.1 DESCRIPTION OF THE MARK III CONTAINMENT AND RELATED SYSTEMS

4.1.1 Reactor and Primary System

The nuclear steam supply system and associated safety systems used in the Mark III plants are very similar in concept and implementation to the reactor system described in Section 3.1 for the Mark II plant. The most recent GE BWR model--known as BWR/6--is used exclusively in the Mark III plants (earlier models BWR/4 and BWR/5 are used in the Mark II plants). BWR/6 is the largest of the GE BWR model series. The thermal output of the twelve plants listed in Appendix D under the Mark III classification ranges from 2894 to 3835 MWt. The fuel quantity in all the plants is approximately 350,000 lb. Steam generation rates in the various units range up to 16,500,000 lb/hr.

Changes in the steam supply system from earlier BWR models include the new 8x8 fuel bundle, improved jet pumps within the reactor vessel, improved steam separators, additional fuel bundles, enhanced ECCS performance, and reduced fuel duty to 13 kW/ft (see Table 3-1 for the salient features of all six BWR models). The primary difference associated with the BWR/6 plants is the use of the Mark III containment. The drywell in the Mark III provides a spacious area for installation of the primary system components such as the recirculation pumps and steam piping. An objective of both the BWR/6 and the Mark III containment designs has been to facilitate licensing of new plants.

4.1.2 Primary Containment and Suppression Pool

Two types of construction have been used for the Mark III containments. Each of these is appropriate for a structure approximately 120 ft in diameter and 185 ft in height (see Figure 4-1). The first employs steel reinforced concrete, a vertical cylinder, a hemispherical dome, a flat base, and a steel liner for the primary containment. The volume of this structure as listed in Appendix E is roughly 1,800,000 ft^3. The Grand Gulf 1 and 2, the Clinton 1 and 2, and the Skagit 1 and 2 plants use this type of construction. The thickness of the heavily reinforced concrete wall is approximately 4 ft and that of the basemat 10 ft. The liner is a relatively

thin welded steel plate, 1/4 to 1/2 inches thick and carries no pressure load. The steel liner forms the leakage barrier while the concrete provides biological shielding and structural strength. The type of concrete used in the plants varies with the composition of the aggregates readily available nearby. Limestone and basalt are commonly used for the coarse aggregate that constitutes the majority of the concrete volume and mass. Products generated during the hot decomposition of concrete are strongly dependent on the type of aggregate present.

The second type of primary containment construction used for the Mark III plants consists of a free-standing steel cylinder and shallow steel dome with a steel-reinforced-concrete base. The base is also covered with a thin steel liner, and the steel cylinder is approximately 1-1/2 inches thick. The steel containment is not so thick that it requires annealing following field assembly to relieve residual welding-induced stresses. Plants built with these materials include Perry 1 and 2, River Bend 1 and 2, and Hartsville 1 and 2. Some designers consider this latter form of containment construction economical while others do not (Walser, 1980). Either construction must withstand DBAs and external assaults as discussed in Appendix A.

The suppression pool is an annular configuration around the inner, lower perimeter of the primary containment. Water is retained by the weir wall (height approximately 20 ft), and steam discharges into the pool from the drywell through submerged, horizontal vents in the lower drywell wall in the event of a steam system rupture in the drywell. The safety relief valves on the primary reactor system (also used as part of the automatic depressurization system) discharge directly into large pipe headers that terminate at spargers submerged in the suppression pool. The suppression pool volume is typically about 160,000 ft^3, similar to a Mark II plant. Water can be drawn from the pool by several of the ECCSs and injected either into the reactor, the reactor piping, or the containment spray headers at the appropriate pressure. The pool has a large heat sink capability. For example, 100 MWh of thermal energy would raise the temperature of the pool 35°F disregarding any evaporation. The RHR system--one of the Engineered Safety Features--can be used to cool the suppression pool during DBA.

The design pressure of the primary containment is 15 psig at a maximum temperature of about 185°F--a considerably lower design pressure than that of the Mark I and II. The design leakage rate at the design pressure is approximately 0.1 percent of the total containment volume per day. Any leakage would be expected to flow to the secondary

containment. Vacuum relief valves prevent negative pressures from exceeding about 0.8 psi. Vacuum relief--if necessary--flows from the annulus space. The drywell design pressure limits are approximately +30 and -20 psig at 330°F. Valves in the drywell limit the differential pressure to prevent suppression-pool water flowback into the drywell.

The upper pool (above the drywell closure head) facilitates refueling activities and provides shielding above the reactor vessel. It contains approximately 75,000 ft^3 of water. A major portion of this water can be transferred by gravity flow to the suppression pool through redundant lines during a LOCA. This action helps assure that the horizontal vent holes in the drywell wall remain submerged during all possible events.

Some of the advantages of the Mark III containment over the Mark I and II plants are enumerated below:

1. The pedestal supporting the reactor vessel is short. This improves the plant's capacity to resist seismic events. Note also that the change in pedestal configuration alters the space available for the installation of core retention devices below the vessel (in comparison with Mark II plants). The dry pedestal in the Mark III may influence whether special cooling provisions beneath the vessel may be required for severe accident mitigation.

2. The primary containment is protected from primary piping rupture-induced damage since the piping is enclosed by the concrete drywell and steam tunnels. Jet impingement and pipe whip do not threaten the containment structure.

3. Nitrogen inerting is not required because of the large containment volume. This fact facilitates maintenance activities within the containment.

4. The lower design pressure simplifies the construction of a steel containment, thereby offering reduced plant capital cost. The lower design pressure, however, allows less tolerance for error in the set-point pressures for the opening and closing of any special vent features added to the containment for severe accident mitigation.

5. Extra space is available in the drywell for the installation and inspection of equipment.

6. The cylindrical drywell shape is easier to construct than the truncated cone used in the Mark II and the steel lightbulb-shaped structure used in the Mark I.

4.1.3 Secondary Containment

The secondary containment of a Mark III plant is known as the shield building. It is approximately 130 ft in diameter, 195 ft high and made of steel-reinforced-concrete 3 ft thick. It, along with complementary portions of the reactor building, completely encloses the primary containment. A gap that varies in width from 5 to 8 ft separates the primary containment from the shield building. The shield building is an Engineered Safety Feature of the plant. Consequently, it is designed to withstand a combination of loadings described in the DBA set and other external conditions. Wind, snow, flood, missiles and earthquake assaults may be a part of the external loading conditions.

Every Mark III plant is provided with a secondary containment (see Appendix E); but in those few plants using concrete with a steel liner rather than having a free-standing steel primary containment, the concrete functions as the shield building. Any leakage from the secondary containment is limited to 25 percent of the annulus volume per day at the full design pressure. Penetrations of the shield building are the same as those associated with the primary containment. They consist of fluid lines, power wiring, instrumentation and control wiring, ventilation ductwork, instrument tubing, equipment hatches, and personnel hatches.

The Mark III annulus vacuum maintenance system (AVMS) maintains the annulus space at a -5 inch water column pressure (below containment atmosphere) during normal plant operation. It shuts down automatically and closes its penetration through the shield building in the event of a containment isolation signal. The negative pressure is selected to assure that in the event of a LOCA the expansion and heating of the primary containment do not result in an annulus pressure higher than the wetwell.

The SGTS is connected to the annulus space located between the shield building and the primary containment. The SGTS recirculates and filters air within the annulus following an isolation signal. The filtration of particulates and halogens prevents or minimizes radioactive releases to the environment and, thereby, meets the criteria of 10 CFR 100. The SGTS maintains annulus vacuum conditions while performing its intended functions.

Two full-sized SGTSs are provided. Each has the ability to process about 6000 scfm. The filter train consists of a moisture eliminator, an electric heater (to improve iodine absorption), a dust filter, a high-efficiency particulate filter (HEPA), a charcoal filter, and a second HEPA filter to catch charcoal particles swept into the vent stream. A electrically powered fan draws annulus air through the filter train and discharges the air back into the annulus for further use or to the outside environment via the plant stack.

4.1.4 Isolation Devices and Protocol

The primary containment of every Mark III plant has hundreds of penetrations. These penetrations are associated with electrical power cables, control and instrumentation wiring, tubing, equipment hatches, fluid lines, and personnel hatches. Sealing these penetrations against leakage is done in a number of ways and, because the primary containment is an Engineered Safety Feature, double seals or barriers are provided in most instances. The design objective for the containment isolation system is to allow normal and emergency passage of fluids into and out of the primary containment while preserving the capability of the containment to limit the escape of radioactive products released during postulated accidents. Leakage releases must be constrained to meet the plant site boundary dose limits.

The containment isolation systems listed below are actuated automatically in the event of any of the following occurrences and a corresponding signal from the plant's instrumentation network:

1. Low water level in the reactor vessel.
2. High drywell pressure.
3. High temperature in the main line steam space.
4. High radiation in the steam line.
5. High flow in the main steam line.
6. Low steam line pressure at the turbine inlet.
7. High radiation levels in the reactor building ventilation exhaust.

Manual actuation to isolate the containment is also possible from the control room and once initiated goes to completion. The actuation signal causes all containment isolation valves to close in systems not required for emergency shutdown (the same signals activate some of the systems associated with emergency core cooling). If necessary, the fluid lines connected to the emergency systems can be closed manually and separately.

In general, two isolation valves are provided for all fluid lines connected to the reactor pressure boundary or containment atmosphere. One valve is located outside and one inside the primary containment barrier (small instrumentation lines that do not represent an overpressurization threat to the containment are fitted with flow-restricting orifices but not isolation valves). Valves that are pneumatically or electrically actuated are spring-loaded to fail in the closed position for nonemergency systems. The valve closure time and valve leakage rates are such that the site boundary dose criteria of 10 CFR 100 are satisfied during DBAs.

Containment isolation is also dependent on the proper functioning of numerous seals and gaskets around containment penetrations which cannot be welded to the containment itself or to the containment liner for some reason. A variety of elastomeric materials are used for this sealing function. These materials, however, are susceptible to overtemperature, humidity, and radiation. Consequently, their performance is checked periodically when the overall containment leakage rate is measured while the containment is deliberately pressurized. The use of double seals in some instances allows individual units to be tested for leak-tightness. A containment temperature limitation results from the temperature sensitivity of these elastomeric materials.

4.1.5 Long-Term Containment Heat Removal

Postaccident heat removal from the containment of the Mark III plant is accomplished by a water spray system working in conjunction with the RHR system. Sprays are used for atmospheric cooling and the condensation of steam that has not reached the suppression pool. They are an Engineered Safety Feature that complements the containment function. Under accident conditions the primary air ventilation system (which is not an Engineered Safety Feature) is isolated and no longer assists in containment temperature control. Some degree of postaccident ventilation is provided, however, by the combustible gas control and containment purge systems described below.

Figure 4-2 schematically illustrates the main elements of the containment spray system. There are two separate and redundant spray systems, each powered by a separate electrical bus with a separate and dedicated diesel generator set for emergency power. Spray operation is similar to that of the wetwell sprays in the Mark I and II plants. In the Mark III plant though, there are no drywell sprays (nor drywell

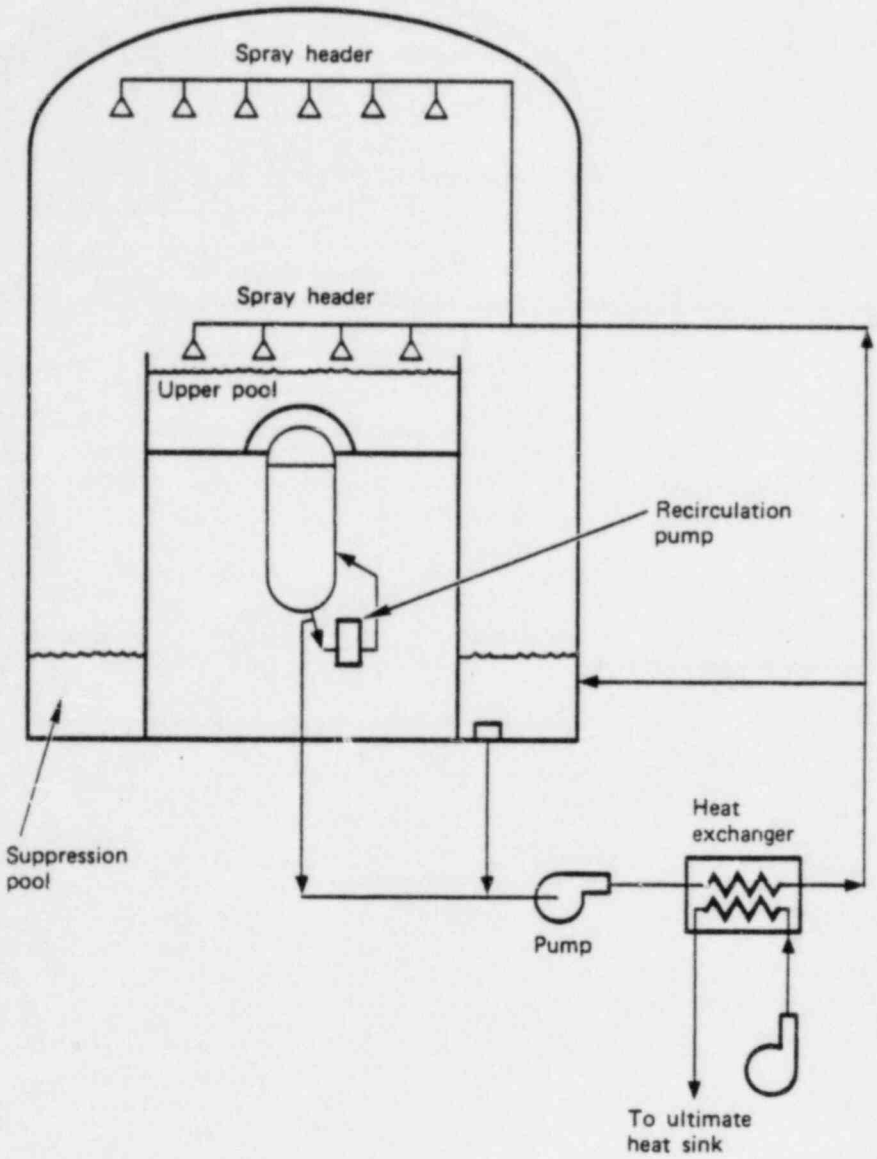


Figure 4-2. Long-term containment heat removal in Mark III plants (redundant system not shown).

emergency sump for return water collection). Two independent spray headers are located high in the primary containment dome. Another header is positioned immediately above the upper water pool. Water is drawn from the suppression pool sump by either system and delivered at a design rate of 5000 to 7000 gal/min to the spray headers. Spray water returns to the suppression pool for cooling and reuse.

The water sprays remove energy from the primary containment by condensing water vapor released during an accident. In this way the sprays cause energy to be conveyed to the suppression pool. The RHR system then carries energy from the pool water to the plant's ultimate heat sink. Activation of the spray system is initiated by the plant operator.

Thermal energy is extracted from the suppression pool water as it passes through the RHR heat exchangers en route to the sprays. The capacity of each of two independent exchangers is about 35,000,000 to 50,000,000 Btu/hr (14.6 MWt maximum). Service water flows through the units at about 5000 to 9000 gal/min and delivers energy to the ultimate heat sink. Electrically driven pumps power the service water system. Use of the spray system can hold the containment temperature and pressure below the design values of 185°F and 15 psig during the most severe LOCA conditions.

The 125,000 ft^3 of concrete and 4500 ft^3 of steel in the containment structure represent a sizeable heat sink--about 93 MWt to 185°F--but will heat up too slowly to be useful during most severe accident scenarios.

4.1.6 Combustible Gas Control

Hydrogen gas may be released to the drywell in the event of a LOCA within the Mark III containment. The gas is a reaction product from the oxidation of Zircaloy fuel cladding (or other metals) in a high-temperature steam environment ($2H_2O + Zr \longrightarrow ZrO_2 + 2H_2$ + energy). The amount of hydrogen that may be produced is shown in Figure 3-7 as a function of the Zircaloy fraction reacted. Hydrogen presents a conflagration or even an explosive danger to the containment wherever it accumulates. Ignition of appreciable quantities can be sufficient to cause loss of containment integrity (Appendix A discusses containment failure modes). High-temperature core debris materials or electrical equipment may serve as uncontrollable hydrogen ignition sources. The design objective of the combustible gas control system is to prevent hydrogen burning or detonating in the Mark III containment by maintaining hydrogen concentrations below the ~4 percent ignition limit. The combustible gas control system

is an Engineered Safety Feature, so it fully redundant and designed to withstand DBA conditions.

The hydrogen control system consists of three elements that can be actuated in stages depending on the rate at which hydrogen is released to the drywell. The three independent actions are:

1. Drywell hydrogen mixing throughout the atmosphere.
2. Hydrogen/oxygen recombining.
3. Containment purging.

Electrical power is needed for operation of each of these hydrogen controls, though some drywell mixing will occur by natural convection.

The drywell hydrogen mixing system uses blowers to force air from the upper portion of the containment into the drywell. This is done with sufficient force to cause circulation of drywell gases through the suppression pool vents and water. In some cases, mixing is able to reduce the highest hydrogen/air ratio to a value less than 0.04, the lower flammability limit. This capability is constrained of course by the net amount of air in the containment. Larger hydrogen quantities require the hydrogen recombining system to be activated.

Two redundant hydrogen recombining systems process air from the drywell atmosphere. Each system forces gas from the drywell through a reaction tube heated by redundant radiant heaters where the temperature is raised sufficiently high to cause hydrogen and oxygen to combine to form water without the presence of an open flame. The hydrogen-free effluent is cooled and returned to the containment above the suppression pool. If this action plus mixing is unable to hold the hydrogen/air ratio below 0.04, then the containment hydrogen purge system is activated.

The containment hydrogen purge system is capable--through either of two redundant systems--to forcibly draw gas from the primary containment and discharge it to the annulus between the primary containment and the shield building. Once in the annulus, the gas can be processed by the SGTS prior to being exhausted to the external environment through the plant stack or returned to the annulus. A small air compressor is used to return clean outside air to the containment as a make-up for that removed to limit hydrogen concentration.

4.2 DOMINANT FAILURE MODES AND ACCIDENT SEQUENCES OF THE MARK III CONTAINMENT

A PRA was prepared for the Grand Gulf 1 plant, a Mark III containment, as part of the Reactor Safety Study Methodology Applications Program (RSSMAP). This plant utilizes the Mark III containment. The RSSMAP was an SNL effort (Hatch, 1981) and was completed well after the RSS, though essentially the same general methodology is used. However, while the RSS addressed a generic BWR represented by the Peach Bottom plant (i.e., a Mark I, 1065-MWe, model BWR/4), the Grand Gulf efforts focused on a 1250-MWe, BWR/6 Mark III plant. Only internal events were considered as accident initiators in these studies.

The RSSMAP was conducted in order to apply RSS methods to other reactor types. Four individual RSSMAPs were conducted: one each for Mark III and ice condenser, and two large, dry containments. The objectives were to (1) identify risk-dominant accidents, (2) compare accidents identified with those of the RSS, and (3) identify design differences that would have a significant impact on risk. Unlike the RSS, detailed fault trees were not used to identify all failure modes, rather a "survey and analysis" technique was employed to identify likely failure modes. The MARCH (Wooten, 1980) and CORRAL (Burian, 1977) codes were used to determine which accident sequences result in core melt and the subsequent containment response. No consequence analysis was performed in the RSSMAP work. The predicted total core-melt frequency for Grand Gulf using the RSSMAP approach is approximately the same as that characterizing the Peach Bottom plant as revealed in the RSS.

This section summarizes accident sequence frequencies for the Mark III plant. RSSMAP results are shown as an indication of conditions created by the Mark III plants in operation or under construction (Appendix D lists twelve plants in this category). It should be noted that the GESSAR II PRA prepared by the GE Company has as its focus a Mark III containment. It was not considered in this work because the design is still undergoing review by the NRC and some aspects are proprietary.

4.2.1 Dominant Accident Sequences and Failure Modes

Twenty-two accident sequences leading to core melt were revealed by the Grand Gulf RSSMAP. Twelve of the 22 were LOCA-related whereas the balance were the result of various transient initiators. The influence of the ATWS-3A-Fix on transient events is not known because it was not included in the PRA. With respect to the frequency of occurrence, the

dominant core-melt accident sequences are the transient events that are accompanied by failure of the RHR system. Approximately 65 percent of the total frequency of core melt is due to this type of sequence. Table 4-2 shows the four main groupings of accident sequences and their relative contributions to core-melt incidents. Table 4-4 explains the accident nomenclature used in Table 4-3. The RSSMAP-projected overall frequency of accidents leading to core-melt is 3.7×10^{-5} per reactor year.

The containment event tree used with each of the core-melt accidents in the RSSMAP is shown in Figure 4-3 (note that the Greek letters used to designate the various containment failure modes in the figure--and therefore in other parts of Sections 4.2 and 4.3--are not consistent with those used in the RSS). Branch point probabilities are associated with every branch of the tree depending on the type of accident under consideration. For example, it was assumed in the RSSMAP that for an SI-type sequence there was a one percent chance ($\alpha = 0.01$) that the first branch point would lead to a steam-explosion-induced containment failure.

Similarly, probabilities were applied to all other branch points for all accident sequences. Depending on whether the containment failure was of the α, β, γ or δ type and depending on the timing of the containment failure, the potential radioactive release from the plant was classified as one of four types. These release types, called "release categories," range downward in severity from Category 1 to Category 4 and are the same categories as used in the RSS. Table 4-5 shows the release categories and their frequency of occurrence for each of the dominant accident sequences. The nature of the release category--as taken from the RSS--is also explained in Table 4-5.

TABLE 4-2. DOMINANT GRAND GULF ACCIDENT SEQUENCE GROUPS

Accident Group	Percent Contribution to Core-Melt Frequency	Applicable Sequences
1. Transients with loss of RHR	65	$T_1PQI/T_{23}PQ$ $T_1QW/T_{23}QW$
2. LOCAs with loss of RHR	12	SI
3. Anticipated transients without scram	15	$T_{23}C$
4. Transients and LOCAs with loss of ECCS	6	$T_1PQE/T_{23}PQE$ T_1QUV
5. Other	2	
Total	100	

Note: External events not included.

TABLE 6-3. GRAND GULF DOMINANT CORE-MELT ACCIDENT SEQUENCES AND THEIR FREQUENCY PER YEAR

Accident Sequence	BWR Core-Melt Release Category* and Containment Failure Type				Percent of Total Frequency
	1 (α)	2 (δ)	3 (γ)	3 (β, δ)	
T_1PQI	1.6×10^{-8}	1.6×10^{-6}			4.7
$T_{23}PQI$	3.7×10^{-8}	3.7×10^{-6}			10
T_1PQE			1.2×10^{-7}	1.2×10^{-7}	0.6
$T_{23}PQE$			2.7×10^{-7}	2.7×10^{-7}	1.5
SI	4.6×10^{-8}	4.6×10^{-6}			12
T_1QW		6.2×10^{-6}			1.6
$T_{23}QW$		1.2×10^{-5}			32
$T_{23}C$		5.4×10^{-6}			15
T_1QUV			7.5×10^{-7}	7.5×10^{-7}	4.0
Total	1.1×10^{-7}	3.4×10^{-5}	1.2×10^{-6}	1.4×10^{-6}	
			Grand Total:	3.7×10^{-5}	

*Release categories are the same as in the RSS but α, δ, γ, β are not; see Table 4-5.

Source: Hatch, 1981 (Figure 6-1)

α = In-Vessel Steam Explosion
β = Containment Leakage
γ = Overpressure: Release to Reactor Building
δ = Overpressure: Release to Atmosphere

External events not included.

TABLE 4-4. KEY TO GRAND GULF (MARK III) SEQUENCE SYMBOLS

Initiating Events

T_1	Loss of off-site power transient
T_{23}	Any other transient which requires emergency reactor shutdown
S	Small LOCA (break area <1 ft^2)

System, Component or Functional Failures

C	Failure to render reactor subcritical
E	ECCS failure
I	RHR failure after a LOCA (including transient-induced)
P	Failure of safety/relief valve to reseat
Q	Failure of the power conversion system
U	Failure of the HPCS and RCIC
V	Failure of the LPCS
W	RHR failure after a transient

TABLE 4-5. BWR RADIOACTIVE RELEASE CATEGORIES

1. Core meltdown followed by an in-vessel steam explosion. Within about 1/2 hour of the explosion-induced failure of the containment, 40 percent of the iodines and alkali metals would leave the containment. The rate of energy release would be high. This category also includes some sequences with overpressure failure prior to core melting.

2. Containment overpressure due to RHR failure precedes core melt. Direct release of radioactivity to the atmosphere would occur without significant retention over a period of 3 hr. Approximately 90 percent of the iodines and 50 percent of the alkali metals would escape.

3. Containment overpressure due to failure to shut down the reactor precedes core melt or it follows a molten core-concrete interaction. About 10 percent of the iodines and 10 percent of the alkali metals would be released. Some retention in the suppression pool would occur.

4. Core meltdown is followed by sufficient containment leakage to prevent overpressure failure. The quantity of radioactivity release would be reduced by retention in the iodines, and 0.5 percent of the alkali metals would be released over a 2-hr period.

5. Only the radioactivity contained within the gap between the fuel pellets and cladding is released to the containment. The core would not melt, and containment leakage would be small. Minimum required Engineered Safety Features would function satisfactorily. A filtered release would occur over a 5-hr period.

Source: NRC, 1975 (Appendix VI).

4.3 BENEFITS FROM MITIGATION SYSTEM INSTALLATIONS

The ideal or hypothetical benefit in risk reduction from the introduction of mitigation systems can be computed based on the frequency of core damage accidents, the conditional probability of containment failure, and their associated consequences. The procedure for doing this was described in Section 2.2. The techniques discussed there will be applied in this section to the Grand Gulf plant. The benefit of each mitigation system will be expressed as a risk reduction factor. This approach is used because the PRA for the Mark III containment does not include external events and has not

been reviewed in the same detail as those for other plants. Several generic mitigation systems will be suggested with each directed at preventing a certain type of containment failure (and subsequent radioactive release following a core-melt accident). Specific mitigation systems that may fully or partially meet ideal requirements are discussed in Section 4.5.

Because the RSSMAP did not compute the consequences of severe accidents, reference is made to the RSS for consequence information. The radioactive release associated with five BWR release categories was quantified in the RSS. These five are shown in Table 4-6, and four of them apply to the Grand Gulf plant (see Table 4-4). Table 4-5 describes the nature of the four release categories relevant to the Grand Gulf plant. Recall that these categories are only a consequence of internal events. The influence of external events on Mark III plants has not been quantified to date.

Table 4-7 lists several mitigation systems and the release categories that may be precluded by their presence in the Grand Gulf plant. Since the RSSMAP concluded with only four release categories in total, it is not possible to offer numerous hypothetical steps for plant improvements. The value of the mitigation system is calculated as a risk reduction factor. The risk reduction factor is obtained by multiplying the frequencies found in Table 4-3 by their corresponding consequences found in Table 4-6. These risk reduction factors are shown in Table 4-7 for acute fatalities and man-rem (which corresponds to latent fatalities).

Examination of Table 4-7 shows that the elimination of release Category 2, which requires additional containment heat removal or a filtered vent and concrete protection system, has the highest risk reduction potential (factors of 164 and 33 for acute fatalities and man-rem respectively). The other factors are negligible. This obvious conclusion occurs because the frequency of Category 2 events is an order of magnitude greater than the others, and the corresponding consequences (in terms of acute fatalities) are much greater.

4.4 MITIGATION OPPORTUNITIES FOR MARK III CONTAINMENTS

The PRA applicable to the Grand Gulf Mark III plant served to identify risks associated with severe accidents characterized by a BWR core melt. The frequency of core-melt incidents is about 3.7×10^{-5} events per year at Grand Gulf (internal events only). The risks associated with radioactive releases from the plant are caused by containment

TABLE 4-6. AVERAGE CONSEQUENCES OF INDIVIDUAL BWR RELEASE CATEGORIES

BWR Release Category	Man-Rem (10^6)	Acute Fatalities
1	2.2	1.7
2	1.8	48
3	0.89	3.0
4	0.42	3.9
5	0.19	1.1

Source: NRC, 1975, draft version.

TABLE 4-7. RISK REDUCTION FACTOR FOR GRAND GULF SEVERE ACCIDENT CONSEQUENCE MITIGATION (RATIO OF RISK WITHOUT IMPROVEMENT TO RISK WITH IMPROVEMENT)

Containment Failure Mode Prevented	Release Category* Eliminated	Acute Fatalities	Man-Rem
α - Missile shield resistant to in-vessel steam explosion consequences	1	1.0	1.0
δ - Containment heat removal system or a filtered vent and concret protection system	2	164	33.0
γ - Filtered vent and concrete protection against strong heating	3	1.0	1.01
β,δ - Scrubbing or filtering of the secondary containment atmosphere	4	1.0	1.01

*See Table 4-4 for Release Category frequency and Table 4-6 for consequences.

failure. In the Grand Gulf plant the failures were primarily due to overpressurization by steam or noncondensible gases.

Effective mitigation of severe accident consequences in Mark III plants requires that containment integrity be maintained under all accident scenarios. Several systems are proposed in this section that have the potential of protecting the containment over both short- and long-term periods. The criteria under which mitigation systems must perform and their major functional features are discussed. Criteria are expressed in terms of assumptions about the availability of plant Engineered Safety Features and the types of molten core/containment interactions to be accommodated. Primary conditions are developed to generate the cumulative requirements described in Chapter 2. Candidate mitigation systems are briefly described, and references are provided as a guide to additional information and design details. No effort is made to size the mitigation systems nor to select specific pieces of equipment. The emphasis is on identifying generic mitigation systems capable of addressing the dominant threats to the Mark III containment barrier.

4.4.1 Cumulative Requirements for Mark III Mitigation

The dominant accident sequences leading to containment failure in the Mark III plant involve various failures in the plant's normal and emergency protection systems. The conservative approach taken for the selection of candidate mitigation systems is to assume that all the failures associated with the dominant accident sequences are present. If the mitigation system can be shown to operate reliably and effectively under this combination of conditions, it should work as well in less stringent environments.

The cumulative listing of plant conditions present during periods when mitigation may be required is found in Section 5.4.1. Basically, the list tabulates failures of the various containment heat removal systems as well as the presence of a fully saturated suppression pool. Furthermore, station blackout conditions are assumed to be present and no operator actions are expected following core melt. Large quantities of hydrogen gas may be present in the containment.

The plant's ultimate heat sink is assumed to remain available throughout the severe accident. This is an important assumption because a primary objective of the proposed mitigation system is to transfer heat from the containment to it (as opposed to directing the heat to some other sink such

as the atmosphere). However, emergency service water systems having access to the ultimate heat sink are not functional.

4.4.2 Criteria for Mitigation Choices

A number of design, installation, operational, and maintenance criteria should be satisfied by mitigation systems selected for use in the Mark III plant. These criteria, or ground rules, should be considered in conjunction with post-core-melt conditions and the other conditions described in the subsection above (i.e., those pertaining to the availability and state of plant Engineered Safety Features). The intention in doing so is to create mitigation systems fully responsive to containment threats.

The ground rules or criteria relevant for the Mark III plant are essentially the same as those proposed for the Mark II in Section 3.4.2. The criteria as described there emphasize conservative design factors, passive operation, resistance to the severe accident environment, and noninterference with Engineered Safety Features.

4.4.3 Potential Severe Accident Conditions to be Mitigated

The assumed state of the plant safety systems as described above and the presence of a core-melt accident combine to produce a number of containment challenges. These threats include and are in addition to the direct causes of containment failure as determined through PRA techniques. The mitigation systems must address these conditions if the anticipated benefits in terms of risk reduction are to be achieved. The conditions are essentially the same as those applicable to a Mark I or Mark II plant. However, because the Mark III containment is not inerted, the presence of hydrogen is of greater concern. Physical events potentially a part of the severe accident sequence include:

1. Rapid steam generation by an ATWS event.

2. Decomposition of concrete by hot core debris. The release of core debris into the pedestal region would result in melting and gasification of concrete constituents.

3. Rapid pressure rise accompanying the initial release of core debris into the containment. Water may be present in the pedestal as a result of an earlier LOCA. Steam will be generated at a rapid rate if hot debris drops into a water pool.

4. Ex-vessel combustible gas production. The chemical reduction of water vapor and carbon dioxide to hydrogen and carbon monoxide is possible if hot steel and/or Zircaloy are present. Hydrogen produced in this manner would add to any formed in the PRV.

5. Continuous production of decay heat in core debris.

Several occurrences that are known to be possible under other circumstances are excluded from the above set because the PRA did not identify them as a part of dominant accident sequences. Conditions in this excluded category include:

1. In-vessel steam explosion.
2. Ex-vessel steam explosions.
3. Containment-threatening missiles.

4.4.4 Cumulative List of Mitigation Functions to be Performed

A review of the likely severe accident conditions and the ground rules applicable to the mitigation methods leads to a selection of the primary functions sought in candidate mitigation systems. Extensive additional efforts would be required in order to convert the functions into the design of systems adaptable to the Mark III plant. The primary thrust of the functions is to preclude the dominant threats to the containment in terms of gradual overpressurization or hydrogen combustion. Furthermore, functions are proposed here that are intended to manage subsequent containment threats if these primary challenges are alleviated. The crucial functions are:

1. Management of hydrogen production and possible subsequent combustion from in-vessel or ex-vessel metal/water reactions.

2. Long-term containment heat removal. The heat removal systems must transfer decay, sensible, latent, and chemical-reaction-generated heat to the ultimate heat sink. This action serves to prevent overpressurization of the containment.

3. Control and confinement of the path of the ex-vessel core debris with ultimate deposition in a localized region. The final resting position of the core debris must be thermally connected to

the containment heat removal system and isolated from concrete structures.

4. Relief of both overpressure and underpressure conditions.

5. Thermal protection of containment penetrations.

4.4.5 Candidate Mitigation Systems for Mark III Containments

Mitigation systems appropriate for use in Mark III containments have been discussed in Hammond (1982) Swanson (1983) Murfin (1980) and Ahmad (1983), and are reviewed briefly in the companion Task 2 report. The mitigation methods considered include filtered-vent systems, hydrogen control systems, containment heat removal, and core retention systems.

A number of different filtered-vent options of general applicability have been proposed (Murfin, 1980; Levy, 1981; Ahmad, 1983; and Reilly, 1982). Designs suitable for use in a Mark III containment have been examined in Levy (1981). Calculations for the case of a release of 5000 lb of hydrogen over 30 min indicate that a vent rate of 32 lb/s (19,400 cfm at 33 psia) should be sufficient in a system derived from the existing SGTS (Levy, 1981). Other designs (Harper, 1982) will accommodate gas flows as large as 400,000 cfm. Complete Zircaloy oxidation would create about 5000 lb of hydrogen.

Unfiltered venting can be considered in the early phase of an ATWS event. Steam production rates may correspond to a heat generation rate as large as 30 percent of full power. Prior to the initiation of core degradation (when steaming rates are high), the steam is uncontaminated and can be released without filtering or scrubbing. If this venting action is taken, then vacuum conditions may develop subsequent to vent closure. Subatmospheric conditions result from steam condensation in the containment. Vacuum breakers would be needed to allow clean outside air back into the containment. Unused containment penetrations may be available for the installation of vents and vacuum breakers.

Hydrogen control systems proposed for use in Mark III containments include spark ignition systems, recombiners, glow plugs, sprays, and foams (Murfin, 1980; Levy, 1981). Some (Levy, 1981) regard the burning of hydrogen as difficult to control properly; they believe that hydrogen burning equipment will require extensive development. The use of spark ignition systems in Mark III containments has been questioned (Levy, 1981) because it is felt that these systems

will not operate reliably in sprays, although their use has been implemented in plants where sprays may be in operation. It is not clear what action will be taken if present studies of igniters in sprays show them to be less effective than desired. Sprays and foams (Murfin, 1980; Levy, 1981; and Reilly, 1982) have been proposed as hydrogen control systems of general applicability to all types of reactor systems.

While inerting the atmosphere of a large containment eliminates the hydrogen combustion problems, it is undesirable during normal operation because service access is complicated (Levy, 1981). Inerting a large containment after the accident requires large equipment since inerting must be done quickly and may add to the already difficult overpressurization problem (Levy, 1981). Heat pipes, large single condenser systems, and water sprays might be used to provide upper containment heat removal (Ahmad, 1983), but suppression pool cooling is a must.

Levy (1981) contends that core retention systems are of no value in Mark III plants without inerting because the containment will fail from hydrogen generated in the early stages of the accident. A flooded cavity is recommended as a substitute because the water will assist in the dissipation of decay heat and delay the time for basemat penetration. Nevertheless, if a core retention system is needed, it appears that the water-cooled crucible installed in a tunnel directly below the vessel (Hammond, 1982) could be used successfully. The flooded pebble bed core retention system (Swanson, 1983) is also a possibility. In new construction, there are a number of other core retention systems that should be considered (e.g., a refractory brick crucible).

Several types of insulation and radiation barriers could be used to protect containment penetrations. Water sprays could also be employed for overtemperature protection.

All the systems proposed above appear feasible in the Mark III containment. Qualifications concerning their use are discussed in greater detail in the companion Task 2 report.

4.5 SUMMARY

The Mark III containment is the most advanced structure to be used with the GE BWR reactors and reflects the design, construction, and operating experiences garnered from the earlier Mark I and II plants. It embodies two major passive safety features: the large suppression pool and the large primary containment volume. These two elements can absorb a larger portion of the energy released during a severe accident

without jeopardizing the containment than can a Mark I or II plant of comparable power. This extends the otherwise relatively short time available for remedial actions. The large containment volume and suppression pool are complemented by the full spectrum of Engineered Safety Features characteristic of other BWR plants. Features relevant to accidents involving core melt include the secondary containment, the SGTS, the containment cooling system, the ultimate heat sink, the containment leakage control system, and the RHR system. The Mark III containment is not inerted with nitrogen as a form of hydrogen combustion/detonation control as is the case in both the smaller-volume Mark I and II containments. Mark III primary containments are built of steel plate, approximately 1-1/2 inches thick. The cylindrical steel shell rests on a thick concrete basemat and is surrounded by a concrete secondary containment.

Data pertaining to the risk associated with Mark III nuclear plant operations were drawn from the Grand Gulf PRA. The possible influence of external events on its analysis of accident sequences, containment performance, and radioactive releases is not addressed in this Mark III PRA. The PRA methodology is essentially that of the RSS. Transient events and LOCAs are used as entry points to event trees. Based on equipment fault tree analysis, the branch point probabilities are assigned to the event tree. The various outcomes of the event trees establish the dominant accident sequences involving core melt and the entry points for containment response trees. The outcome of the containment trees establishes in turn the timing and quantity of radioactive release from the plant.

The estimated frequency of core-melt accidents in the Grand Gulf plant is 3.7×10^{-5} per reactor year. Approximately 65 percent of the Grand Gulf core-melt incidents are the result of transient accident initiators, with subsequent failure of the RHR system. The ATWS is the next most common core-melt accident. It contributes 15 percent to core-melt frequency. With respect to the type of containment failure, 92 percent of the core-melt incidents lead to a δ-type failure (i.e., rupture due to steam and/or noncondensible gas generation). The BWR 2 release category from the RSS is characteristic of this type of failure, and roughly 1.8×10^{6} man-rem release is the consequence with a corresponding risk of 61 man-rem per year.

Distinct mitigation systems are proposed for the Mark III plant. The value in doing so is based on a risk reduction factor in terms of acute fatalities and man-rem.

Mitigation in a Grand-Gulf-type plant should address slow overpressurization through containment heat removal and concrete protection from ex-vessel core debris. Protection against ATWS events is also necessary. A large capacity containment relief could be used to relieve relatively clean steam from an ATWS event and then reclose prior to core melt. This action should be followed with the use of a highly reliable containment heat removal system involving, for example, water sprays with transfer of energy to the plant's ultimate heat sink. A core retention device would probably be needed later to isolate the core debris from the concrete. This device would also need to be actively cooled. The value associated with taking these mitigation steps is estimated as risk reduction factors of 164 for acute fatalities and 33 for man-rem respectively.

As a part of choosing a mitigation scheme, the question of as-built drywell to containment leakage must be addressed. Presently, the Mark III technical specifications allow 500 cfm at a ΔP of 3 psi. What this means in terms of suppression pool bypass is not clear. Other sequences such as loss of vacuum breakers will have to be factored into value/impact studies.

The risk reduction factor associated with Mark III severe accident consequence mitigation systems ranges up to about 160, depending on the type of containment failure precluded. Further detailed analysis is needed to determine if cost-effective systems can be designed for plants yet to be built. The low-risk values are a reflection of the improved efficacy of the Mark III Engineered Safety Features relative to earlier plants. The results of this improvement are a reduction in both core-melt frequency and radioactive release associated with those containment failures that do occur.

CHAPTER 5. BWR MARK I CONTAINMENT

The Mark I containment for GE BWR plants is the oldest standard containment configuration used for BWRs (Wade, 1974). It has since been replaced in newer plants by the Mark II and III configurations. Appendix D lists 24 U.S. plants with Mark I containments. Nearly every one of the units has been granted an operating license by the NRC. The oldest operating unit--Oyster Creek 1 in Oyster Creek, New Jersey--received its operating license in April 1969. The electrical output of the operating plants ranges from a low of 541 MWe for the Vermont Yankee plant to a high of 1065 MWe at each of the three units at Browns Ferry. Fermi 2, a plant yet to receive its operating license, has an output of 1093 MWe, larger than any other Mark I plant. The Mark I containment design evolved as a result of the increase in BWR power from that exhibited by the pre-Mark plants (i.e., Big Rock Point at 75 MWe and La Crosse at 50 MWe). Figure 5-1 is a sectional view of a typical Mark I containment. It shows as distinguishing features the lightbulb-shaped drywell containing the reactor vessel and the torus (or wetwell) containing the suppression pool of water.

The pressure-suppression type of containment inherent in the Mark I design was selected by plant designers because of the large energy content of the water and steam in the reactor's primary system. The thermal energy stored in the form of pressurized hot water and steam is significantly larger than that in a PWR of comparable thermal output because there is more coolant. This extra fluid is necessitated by the larger BWR pressure vessel (the vessel must accommodate a steam separator and dryer in its upper portion). A dry containment sized to accommodate LOCA conditions would be large and, consequently, expensive to construct. As an alternate approach, steam pressure can be suppressed in a relatively small volume using a water pool. Generally speaking, if all the primary system water flashes to steam in an accident, then roughly 90 percent of it would need to be condensed in the suppression pool to prevent containment overpressurization (Glasstone, 1981). Routing steam and noncondensible gases released during a severe accident through the suppression pool has the advantage of radioactive species scrubbing action as well. The primary containment boundary formed by the drywell, the suppression pool, and the interconnecting duct work is designed to be leak-tight.

The nuclear plant's reactor building completely surrounds the primary containment. Leakage can be collected and processed in the reactor building outside the containment. The relatively small volume of the Mark I containment

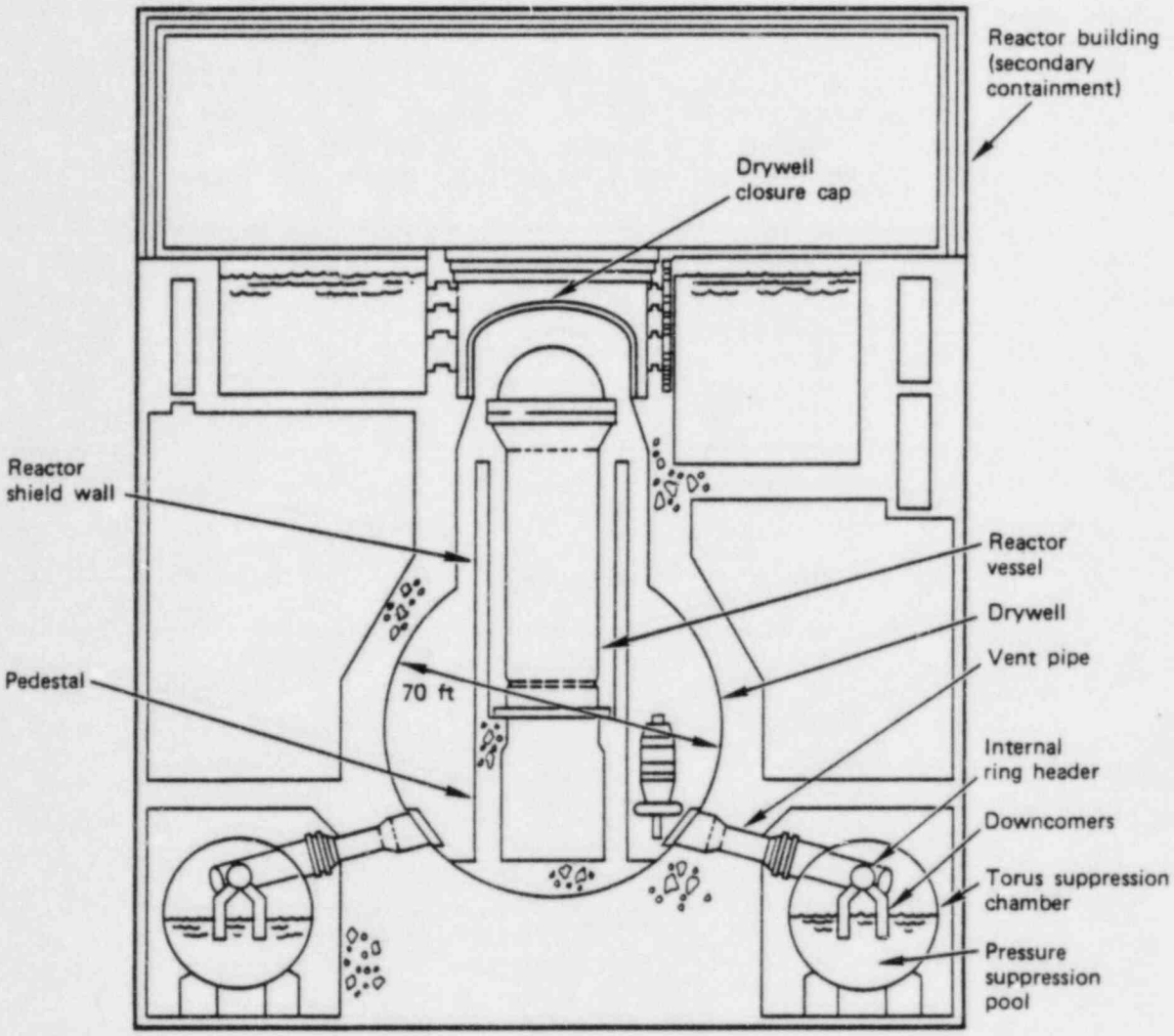


Figure 5-1. Mark I containment.

(approximately 300,000 ft^3) is unable to appreciably dilute hydrogen potentially released to it following a metal/water reaction in the reactor vessel. Such reactions are possible if the fuel cladding overheats due to insufficient coolant flow during a severe accident. Hydrogen burning or even detonating is possible if the hydrogen/air ratio exceeds 0.04. High ratios could be produced if only a few percent of the Zircaloy cladding were oxidized in a very hot steam environment. Consequently, the primary containment is inerted with nitrogen. The oxygen content is reduced to about 5 percent. No routine maintenance is performed while the containment is inerted. High radiation levels preclude normal entry to the drywell during plant operation. If workers need to enter the containment, the reactor is shut down to a hot standby or cold shutdown condition and the atmosphere purged with fresh air.

The Mark I containment is discussed in further detail in Section 5.1. The containment is an Engineered Safety Feature of the plant and, because of this, it is designed to more stringent standards than other plant features. The DBA set provides an envelope of severe accident environments that the Engineering Safety Features must accommodate without adverse radiation release from the plant (see Appendix B). Other Engineering Safety Features directly complement and make possible the successful operation of the containment. Complementary Engineering Safety Features include the containment cooling system, atmospheric control (e.g, nitrogen inerting), isolation features, and the secondary containment. These systems are also discussed in Section 5.1, and reference is made to the DBA conditions they can accommodate as well as other conditions they cannot.

PRA studies have highlighted accident sequences beyond the design basis that can cause containment failure. The dominant accident sequences leading to failure are identified in Section 5.2. Mitigation systems are then proposed in Sections 5.3 and 5.4 that may be able to alleviate--at least to some extent--the consequences of severe accidents (i.e., Class 9 accidents). The degree to which this is theoretically possible is estimated below along with the value of any risk reduction. The calculation of quantitative change in risk is made through modifications of the PRA results. PRA study results are available for the Nine Mile Point 1, Millstone 1, Browns Ferry 1, and Peach Bottom plants. It is assumed in Section 5.3 that certain containment failure modes can be wholly eliminated through the action of selected ideal mitigation systems (later tasks will estimate the actual extent to which real-world mitigation systems can match this ideal performance). The value of risk reduction is computed on the basis of the change in

man-rem caused by eliminating various release categories. Suggestions are made for further detailed study of mitigation systems in future tasks (Section 5.5).

5.1 DESCRIPTION OF THE MARK I CONTAINMENT AND RELATED SYSTEMS

5.1.1 Reactor and Primary System

The nuclear steam supply and safety systems for the Mark I plant are very similar to those described in Section 3.1 for the Mark II plants. GE models BWR/1 through BWR/4 are used with the Mark I containments. Model BWR/4 is used in some Mark II plants as well (see Table 3-1). The thermal output of the 24 Mark I plants listed in Appendix D ranges up to 3293 MWt (i.e., Fermi 2). The smallest unit--Vermont Yankee--has a thermal output of 1593 MWt. Steam delivery rates range from 6,000,000 to 14,000,000 lb/hr. Fuel weight (UO_2) ranges from 180,000 to 370,000 lb, with corresponding fuel cladding (Zircaloy) masses of 63,000 and 130,000 lb respectively.

The differences in the various BWR models include water volume, coolant flowrate, fuel assembly arrangement, reactor vessel size, and ECCSs. Some of the specific parameters applicable to a representative plant with model BWR/4 are listed in Table 5-1.

TABLE 5-1. MARK I PLANT NUCLEAR SYSTEM BOTTOM PLANT (BWR/4)

Parameters	Values
Rated power (MW)	3293
Steam flowrate (lb/hr)	13.4×10^6
Number of fuel assemblies	764
Core UO_2 weight (lb)	370,000
Core zirconium weight (lb)	127,000
Core diameter (in)	187
Core height (in)	144
Number of control rods	183
Number of jet pumps	20
Core spray system flowrate (gpm/loop)	6250 @ 122 psid (2 loops)
HPCI flowrate (gpm)	5000
LPCI flowrate (gpm/pump)	10,000 (2 pumps)
RCIC flowrate (gpm)	616 @ 1120 psid
RHR heat exchanger duty (BTU/hr/exchanger)	70×10^6 (4 exchangers)

5.1.2 Primary Containment and Suppression Pool

The Mark I primary containment (see Figure 5-1) consists of the lightbulb-shaped drywell and the torus-shaped wetwell containing the suppression pool of water. The lightbulb shape is a result of two design objectives. First, a small-diameter removable closure cap above the reactor is desirable; second, space is needed around the bottom of the reactor vessel for recirculation pumps. The light bulb configuration meets both of these criteria. The torus shape for the wetwell provides a large surface area for introducing the multiple, vertical downcomer lines which route steam into the pool. The spherical portion of the drywell has a diameter of approximately 70 ft and the cylindrical section 33 ft.

Two types of construction have been used for the Mark I containments. The use of steel-reinforced concrete with a steel liner for both the drywell and the wetwell is one construction technique. The Fermi 2 plant is built in this manner. All the other Mark I containments are constructed of heavy steel plate. The design pressure ranges from 35 to 62 psig for the primary containment. The pressure expected to be generated by the most severe DBA is predicted on the basis of analysis and experiments to be less than the design pressure. Provisions are included in the plant to test the leakage rate of the containment at 115 percent of the design pressure as required by federal regulations. The design leak rate is normally 0.5 percent of the containment free volume per day at the full design pressure.

The steel drywell expands and contracts as the temperature within changes as a function of the operational status of the reactor. Consequently, a small air gap exists around it to accommodate this growth. To control yielding of the steel shell by possible jet impingement during a LOCA, the gap is backed by a concrete wall. This concrete also serves as a radiation shield.

The suppression pool contains approximately 120,000 ft^3 of water (900,000 gal). The torus containing the water has a major diameter of about 110 ft and a minor diameter of about 30 ft. This water serves as a large heat sink in the event of a reactor accident. Eight-foot-diameter ducts connect the drywell to the wetwell torus. The large ducts branch through a vent header into multiple 2-ft-diameter downcomers that have their open lower ends submerged in the water pool. Steam released by broken pipe(s) in the drywell is channeled by the ducts and downcomers in the pool. Steam can also be

directed into the pool by separate lines from the safety/relief valves on the reactor's primary system. These valves serve to protect the primary system from excessive pressure and can also serve to reduce vessel pressure quickly. A low system pressure allows low-pressure cooling water to be injected into the vessel. Doing so quickly is important if the high-pressure coolant systems fail following a LOCA. Vacuum breakers allow noncondensibles to return to the drywell from the wetwell. This action prevents backflow of water in the downcomers.

Several Engineered Safety Features are connected to the suppression pool. Both the high- and low-pressure ECCSs draw water from the pool. The containment spray system does likewise (see Figure 5-2, Section 5.1.5). The temperature of the suppression pool water can be controlled through the use of multiple RHR heat exchangers. Water from the pool can be passed through the RHR exchangers and cooled by water circulated through the same units from the plant's ultimate heat sink.

The portion of the primary containment directly below the reactor vessel is dry. Consequently, cooling provisions may need to be added to this area if a core retention/cooling device is provided as an ex-vessel, severe accident mitigation system. The space available for use by new equipment is limited and is dependent on the plant design. The pedestal that supports the reactor vessel, piping, control drives, and multiple recirculation pumps fills a large portion of the bottom of the containment. The primary criterion used in setting the size of the drywell was the need to accommodate these items.

5.1.3 Secondary Containment

The secondary containment of the Mark I plant is the shell of the reactor building. It completely encloses the primary containment which consists of the drywell and wetwell portions. It is normally constructed of steel-reinforced concrete (with no liner). The secondary containment is an Engineered Safety Feature. As such, it is designed to maintain its containment functions despite DBAs and external events. Snow, wind, missiles, floods and earthquakes can create external loadings of varying extent depending on the plant site. In the event radioactivity is released to the reactor building atmosphere, the secondary containment contains equipment to isolate, contain, and assure the filtered, elevated release of the secondary containment atmosphere. This equipment can be tested during normal plant operations.

Several floors supporting plant equipment are located within and attached to the secondary containment. The equipment includes items used for reactor refueling, reactor servicing, high-voltage AC equipment, emergency pumps and electrical gear. Workmen present on these floors are shielded from radiation during normal plant operation by the concrete surrounding the drywell and the refueling pool located above the drywell closure cap.

The design pressure is low, approximately 0.25 psig with a corresponding design leak rate of 100 percent of the secondary containment volume or less per day (see Appendix E). A function of the secondary containment is to provide a collection point for gases leaked or purged from the primary containment. Collection is facilitated by maintaining the secondary containment at a slight negative pressure during normal operations and when isolation conditions are required. This collection role provides an opportunity for gas processing prior to its release to the environment. The requirements of 10 CFR 100 regarding site boundary dose limits encourage the processing of primary containment leakage for removal of radioactive constituents prior to their release through the plant stack. Further details concerning this gas treatment system are found below. Radioactive materials may be released to the secondary containment during a LOCA in the drywell or during purging operations when the primary containment is opened.

The drywell closure head on the primary containment is removed during refueling and other reactor servicing operations. The primary containment function is lost when this occurs. Consequently, the secondary containment serves as the primary containment during these periods and any other times when the primary containment is opened. Sealable personnel and equipment hatches used for access to the reactor building are necessary to assure secondary containment integrity.

5.1.4 Isolation Devices and Protocol

The primary containment of the Mark I plant is pierced by hundreds of penetrations. These penetrations are associated with electrical power, control and instrumentation wiring; ventilation ductwork; piping; instrument tubing; and equipment and personnel hatches. Sealing against leakage is provided in a number of ways and, because the primary containment is an Engineered Safety Feature, double seals or barriers are provided except in special cases. The design objective for the containment isolation system is to allow normal and emergency passage of fluids into and out of the primary

containment while preserving the capability of the containment to limit the escape of radioactive products released during postulated accidents. It is required that releases be constrained to meet the criteria pertaining to plant site boundary dose limits.

The containment isolation systems are actuated automatically in the event of any of the following occurrences and a corresponding signal from the plant's instrumentation network:

1. Low water level of the reactor vessel.
2. High drywell pressure.
3. High radiation level in the reactor building ventilation exhaust.

Manual actuation is also possible from the control room and once action is initiated, the function goes to completion. The actuation signal causes all containment isolation valves to close in systems not required for emergency shutdown (the same signals activate some of the systems associated with emergency core cooling). If necessary, the fluid lines connected to the emergency systems can be closed manually. The primary containment isolation signal also isolates the secondary containment (i.e., the reactor building), shuts down the normal ventilation equipment, and activates the SGTS.

In general, two isolation valves are provided for all fluid lines connected to the reactor pressure boundary or containment atmosphere. One valve is located outside and one inside the primary containment barrier (small instrumentation lines that do not represent an overpressurization threat to the containment are fitted with flow-restricting orifices but not isolation valves). Valves that are pneumatically or electrically actuated are spring-loaded to fail in the closed position for nonemergency systems. The valve closure time and valve leakage rates are such that the site boundary dose criteria of 10 CFR 100 are satisfied during DBAs.

Containment isolation is also dependent on the proper functioning of numerous seals and gaskets around containment penetrations which cannot be welded to the containment itself or to the containment liner. A variety of elastomeric materials are used for this sealing function. The materials, however, are susceptible to temperature, humidity, and radiation. Consequently, their performance is checked periodically when the overall containment leakage rate is measured while the containment is deliberately being pressurized. The use of double seals in some instances allows individual units to be tested for leak-tightness.

The containment temperature limitations are a consequence of the temperature sensitivity of the elastomeric materials.

5.1.5 Long-Term Containment Heat Removal

The primary containment spray system acts in concert with the containment isolation devices to help assure minimal release of radioactivity in the event of a DBA. The spray system is responsible for the control of temperature and pressure in the containment. It is a subsystem of the RHR system and is diagrammed in Figure 5-2. The entire RHR system is designed to be an Engineered Safety Feature of the plant. To meet these safety criteria, the spray system is designed to accommodate all DBA loads (i.e., temperature, pressure, humidity, fluid jet forces, pipe whip); the safe-shutdown earthquake; and internal missiles. The containment ventilation system which controls containment atmospheric temperature during normal plant operation shuts down during accidents. Normally, the atmospheric temperature is held below about 150°F in order to prevent deterioration of wiring insulation, gaskets, sealants, etc. Ventilation during postaccident periods is provided in some degree by the SGTS and purge system described below.

The primary containment spray system consists of two fully redundant trains. Water is drawn from a suction header connected by screened openings to the suppression pool, routed through the RHR heat exchangers, and directed to spray headers and associated nozzles in the wetwell. The spray headers consist of 6- to 12-inch-diameter steel pipes with multiple nozzles positioned along the pipes. The design flow rates of each of the electrically driven spray water pumps is about 10,000 gal/min. Each of the two heat exchangers associated with the RHR system is sized to handle about 70,000,000 Btu/hr. Energy is transferred in the exchangers to water pumped to and from the ultimate heat sink. Steam condensation and heat extraction by the sprays serve to control containment temperature and pressure as well as to maintain the effectiveness of the pool as a steam suppression device. Water sprayed into the drywell flows back into the suppression pool for reuse through the large vent pipes leading to the suppression pool torus.

The wetwell spray system is actuated automatically while a signal initiated by the plant operator is needed to send RHR water to the drywell spray headers. No chemicals are added to the spray water to enhance scrubbing efficiency. Scrubbing is accomplished within the suppression pool. Approximately 135,000 ft^3 of water is available in the suppression pool. In the event that steam condensation and purging

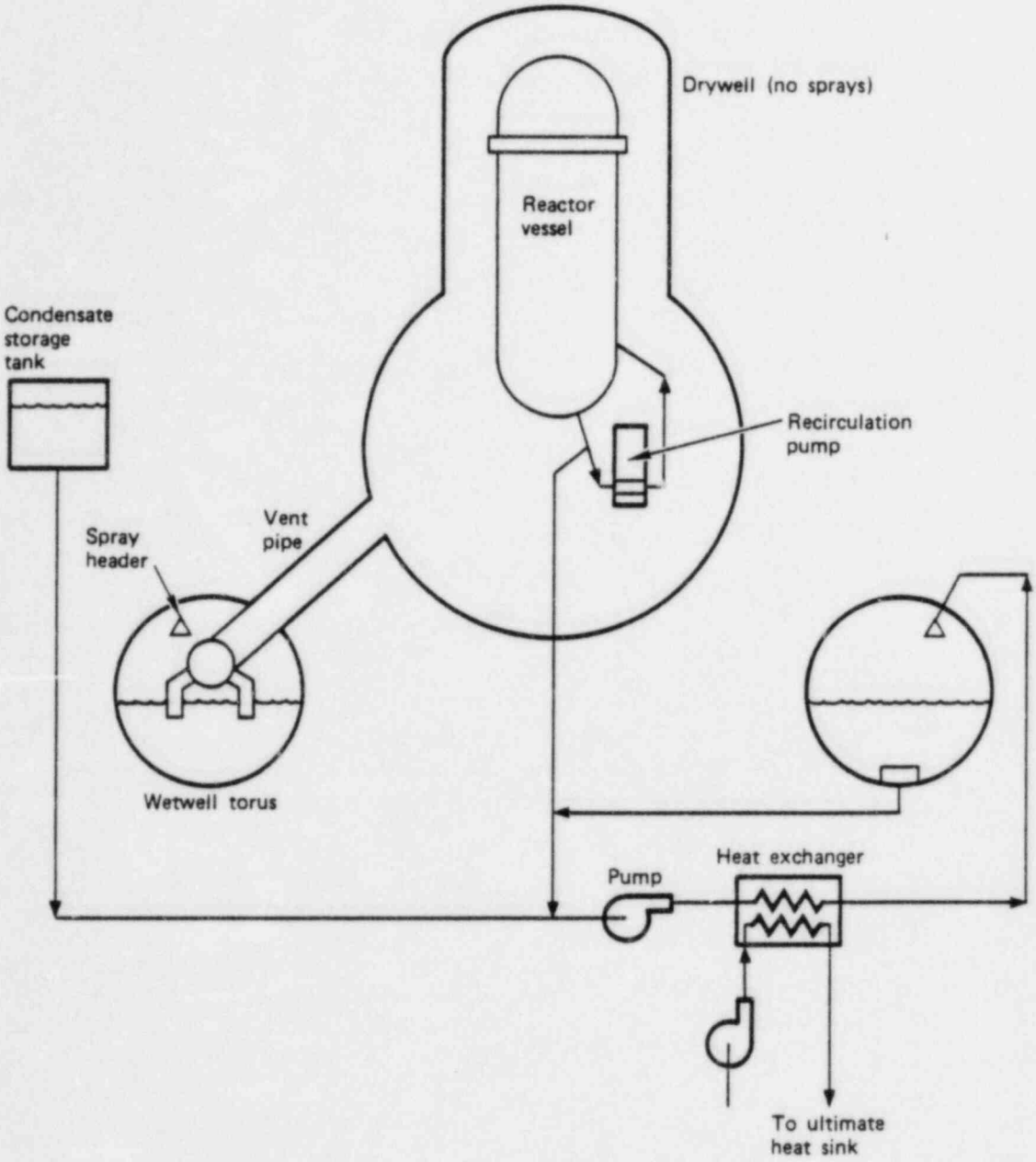


Figure 5-2. Long-term containment heat removal in Mark I plants (redundant system not shown).

cause the primary containment pressure to drop below atmospheric pressure, relief valves allow clean air to enter the wetwell from the reactor building.

5.1.6 Combustible Gas Control

Hydrogen gas may be added to the containment following a LOCA or other events that breach the reactor primary system. Hydrogen is potentially available from several sources, as noted below:

1. Metal-water reactions involving the Zircaloy fuel cladding or other metals in the reactor. The oxidation of Zircaloy can, under ideal conditions, produce thousands of pounds of hydrogen (see Figure 3-7).

2. Radiolytic decomposition of coolant water. This reaction also liberates oxygen.

3. Corrosion of metals and paints in the primary containment. The use of containment sprays may accelerate the corrosion process.

The burning or detonation of hydrogen presents a serious threat to containment integrity during severe accidents (see Section 5.2 and Appendix A for a discussion of containment failure modes). Historically, 10 CFR 50 required that provisions be included in the plant to accommodate hydrogen production from the reaction of 5 percent of the Zircaloy cladding. Since the TMI-2 accident, the potential for larger hydrogen amounts has received widespread attention. As a consequence, inerting of the Mark I containment has been required.

The drywell and wetwell are inerted with nitrogen during power operation of the reactor (i.e., when the electrical output exceeds 5 percent of the rated plant capacity). The nitrogen serves to displace oxygen from the primary containment atmosphere. The concentration of oxygen is controlled to a level below the lower flammability limit. Typically, the oxygen content is held below 5 percent to satisfy this criterion. The use of inerting complicates maintenance activities within the primary containment because of the need for purging and reinerting whenever maintenance activities are conducted. It also represents a potential asphyxiation hazard to workers.

The long-term generation of hydrogen and oxygen by radiolysis is controlled by dilution and recombination. The drywell air-cooling system is used to circulate air within the

primary containment. This mixing action prevents the local accumulation of hydrogen and/or oxygen. An electrically operated recombiner is also present. It heats a small stream of gas drawn through it to a temperature of about 1200°F. At this temperature the hydrogen will react with oxygen in the gas stream to form water vapor. The capacity of the recombiner is low, consistent with the gradual production of hydrogen by radiolysis.

In addition to the mixing and recombiner systems, a third aspect of the primary containment atmospheric control is a purging system. All three relate to hydrogen control and all are Engineered Safety Features. Two full capacity purge systems are available to the plant operator. They can forcefully withdraw gas from the primary containment and deliver it to the reactor building (i.e., to the secondary containment). Once in the secondary containment, the gas can be processed by the SGTS prior to its release to the environment via the plant stack or recycling for further hold-up in the reactor building. The SGTS acts to remove halogens and particulates from the gas. Purging is only used if the other hydrogen control measures are unsuccessful in adequately controlling hydrogen concentration in the primary containment. Additional blowers are present and can be used to return clean outside air to the containment to make up for gas lost during purging.

5.2 DOMINANT FAILURE MODES AND ACCIDENT SEQUENCES OF THE MARK I CONTAINMENT

The threats to Mark I containment integrity were identified and discussed in several PRA studies as well as in the NRC-sponsored SASA program. The Peach Bottom Mark I plant was the representative BWR unit in the seminal RSS (NRC, 1975). The RSS identified BWR containment failure modes, the dominant accident sequences responsible for failure, expected frequency of failure, and the consequences of radioactive releases from the plant. The RSS PRA methodology was used subsequently in the Interim Reliability Evaluation Program (IREP). The IREP was mandated by the Three Mile Island Action Plan (NRC, 1980). It has the objectives of (1) identifying accidents primarily responsible for risk, (2) developing techniques for more intensive PRA exercises, (3) expanding the number of PRA practitioners, and (4) developing PRA procedures applicable to a range of plants. The Browns Ferry Unit 1 (Mark I) plant was included as one of the four nuclear plants analyzed using PRA techniques in the IREP (Mays, 1982). As with the RSS, external events were not considered in the IREP work. Another feature of the

IREP work is that no detailed consequence information was generated. Reference was made instead to the RSS for a description of the various release categories associated with containment failure.

The SASA program involved the investigation of major events that may occur in a nuclear power plant following a number of postulated transient accident initiators. One of the SASA studies focused on a BWR 4/Mark I plant (Yu, 1982) and accidents beyond the design basis. Browns Ferry Unit 1 also served as the model plant for this work. The MARCH thermo-hydraulic computer code (Wooten, 1980) was used extensively by SASA investigators at ORNL in the assessment of Mark I severe accidents. The pertinent accidents were those identified in the RSS as the main contributors to risk at a BWR nuclear plant. A major finding of the SASA program was that containment failure occurred at a significantly earlier time following the accident initiator than expected based on RSS results because of temperature-induced failure of the containment electrical penetration seals. Containment failure time was found to be reduced by 28 to 91 percent depending on the accident sequence. This finding is not necessarily entirely disadvantageous however. Leakage through penetrations may preclude later catastrophic containment failure, and some filtering action would occur in the penetration passages. The results of the SASA were used to substantiate the NRC Emergency Action Level Guidelines (NRC, 1979), but are still being investigated.

The SASA results are not included formally in the discussion below on dominant accident sequences because their quantitative influence on the frequency of containment failure and radioactive release amount was not established. Reference is being made to SASA results to illustrate that the PRA information is subject to updating and revision as accident development insights become more sophisticated. Another exercise showing that the RSS results should be updated for BWR plants was performed by EPRI (Levy, 1981). The EPRI work identifies these specific areas for motivating revisions in RSS results: ATWS, steam explosions, basemat penetration, hydrogen burning, and other improvements. To put the SASA studies in proper perspective, it should be noted that the recent SNL studies are showing that the electrical penetrations may not be as fragile as once thought. Furthermore, the SASA MARCH calculations are based on core/concrete interaction modeling by the INTER code which is known to be poor.

The RSS provided the first comprehensive listing of BWR Mark I accident sequences leading to core melt (the RSS assumed

that all core-melt incidents eventually resulted in containment failure and radioactive release). The results apply to a BWR/4 reactor operating within an inerted Mark I containment. Five radioactive release categories were used to characterize the spectrum of the containment releases following severe accidents. These range downward in seriousness from release category BWR 1 to BWR 5. These categories are detailed in Table 4-5, and the consequences associated with each are shown in Table 4-6 in terms of man-rem and acute fatalities. Each category is characterized by a frequency duration, elevation of release, energy, and radioactive isotope composition.

The dominant BWR Mark I accident sequences (for the Peach Bottom Plant) contributing to release categories are shown in Table 5-2. This information is taken directly from the RSS. The major contributors are transients followed by failure of the RHR system. The nomenclature used in Table 5-2 is explained in Table 5-3, and the containment event tree used for a BWR is shown in the RSS (NRC, 1975). The failure of the greatest probability are those due to overpressurization following a transient event. This point is illustrated in Table 5-2. In this latter table the frequency of containment failure by type is given for each of the four relevant BWR release categories associated with core melt. The largest table entries correspond to an overpressure failure (γ) accompanied by a Category 3 release. The numerical values shown in Table 5-4 differ slightly from those in Table 5-2. This is because the total probability in each category was calculated as proposed by Jaffe (1979) by summing the probabilities of the accident sequences within each category rather than determining it by using the RSS Monte Carlo technique.

The other source of information pertaining to Mark I dominant accident sequences is the IREP studdy (Mays, 1982). As mentioned above, this work focused on the Browns Ferry Unit 1 plant. The results of the IREP work are shown in Table 5-5 with an explanation of the nomenclature found in Table 5-6 Eight dominant sequences were identified, and they are the ones listed in Table 5-5. The containment failure modes and release categories are the same as those used in the RSS.

Three Browns Ferry sequences ($T_UR_BR_A$, TKR_BR_A and $T_UQR_BR_A$) involve transients with subsequent failure of the suppression pool and shutdown cooling systems. Failure of the RHR system eventually results in an inability to pump (due to pump cavitation when drawing on a saturated water source) water back into the reactor. Core dryout and melting then ensue. They do not occur, however, for several hours because of the time necessary to raise the suppression pool temperature to saturation.

TABLE 5-2. BWR DOMINANT ACCIDENT SEQUENCES OF EACH EVENT TREE VS. RELEASE CATEGORY

	RELEASE CATEGORIES				
				Core Melt	No Core Melt
	1	2	3	4	5
LARGE LOCA DOMINANT ACCIDENT SEQUENCES (A)	$AE-\alpha$ 2×10^{-9} $AJ-\alpha$ 1×10^{-10} $AHI-\alpha$ 1×10^{-10} $AI-\alpha$ 1×10^{-10}	$AE-\gamma'$ 3×10^{-8} $AE-\beta$ 1×10^{-8} $AJ-\gamma'$ 2×10^{-9} $AI-\gamma'$ 2×10^{-9} $AHI-\gamma'$ 2×10^{-9}	$AE-\gamma$ 1×10^{-7} $AJ-\gamma$ 1×10^{-8} $AI-\gamma$ 1×10^{-8} $AHI-\gamma$ 1×10^{-8}	$AGJ-\delta$ 6×10^{-11} $AEG-\delta$ 7×10^{-10} $AGHI-\delta$ 6×10^{-1}	A 1×10^{-4}
A PROBABILITIES	8×10^{-9}	6×10^{-8}	2×10^{-7}	2×10^{-8}	1×10^{-4}
SMALL LOCA DOMINANT ACCIDENT SEQUENCES (S_1)	$S_1E-\alpha$ 2×10^{-9} $S_1J-\alpha$ 3×10^{-10} $S_1I-\alpha$ 4×10^{-10} $S_1HI-\alpha$ 4×10^{-10}	$S_1E-\gamma'$ 4×10^{-8} $S_1E-\beta$ 1×10^{-8} $S_1J-\gamma'$ 7×10^{-9} $S_1I-\gamma'$ 7×10^{-9} $S_1HI-\gamma'$ 6×10^{-9}	$SE-\gamma$ 1×10^{-7} $S_1J-\gamma$ 3×10^{-8} $S_1I-\gamma$ 4×10^{-8} $S_1HI-\gamma$ 2×10^{-8} $S_1C-\gamma$ 3×10^{-9}	$S_1GJ-\delta$ 2×10^{-10} $S_1GE-\delta$ 2×10^{-10} $S_1EI-\varepsilon$ 1×10^{-10} $S_1GHI-\delta$ 2×10^{-10}	
S_1 PROBABILITIES	1×10^{-8}	9×10^{-8}	2×10^{-7}	2×10^{-8}	
SMALL LOCA DOMINANT ACCIDENT SEQUENCES (S_2)	$S_2J-\alpha$ 1×10^{-9} $S_2I-\alpha$ 1×10^{-9} $S_2HI-\alpha$ 1×10^{-9} $S_2E-\alpha$ 5×10^{-10}	$S_2E-\gamma'$ 1×10^{-8} $S_2E-\beta$ 4×10^{-9} $S_2J-\gamma'$ 2×10^{-8} $S_2I-\gamma'$ 2×10^{-8} $S_2HI-\gamma'$ 2×10^{-8}	$S_2E-\gamma$ 4×10^{-8} $S_2J-\gamma$ 8×10^{-8} $S_2I-\gamma$ 9×10^{-8} $S_2HI-\gamma$ 9×10^{-8} $S_2C-\gamma$ 8×10^{-9}	$S_2CG-\delta$ 6×10^{-11} $S_2GHI-\delta$ 6×10^{-10} $S_2EG-\delta$ 3×10^{-10} $S_2GJ-\delta$ 6×10^{-10} $S_2GI-\delta$ 2×10^{-10}	
S_2 PROBABILITIES	2×10^{-8}	1×10^{-7}	4×10^{-7}	4×10^{-8}	

TABLE 5-2. BWR DOMINANT ACCIDENT SEQUENCES OF EACH EVENT TREE VS. RELEASE CATEGORY (CONCLUDED)

	RELEASE CATEGORIES				
				Core Melt	No Core Melt
	1	2	3	4	5
TRANSIENT DOMINANT ACCIDENT SEQUENCES (T)	TW-α 2×10^{-7} TC-α 1×10^{-7} TQUV-α 5×10^{-9}	TW-γ' 3×10^{-6} TQUV-γ' 8×10^{-8}	TW-γ 1×10^{-5} TC-γ 1×10^{-5} TQUV-γ 4×10^{-7}		
T PROBABILITIES	1×10^{-6}	6×10^{-6}	2×10^{-5}	2×10^{-6}	
PRESSURE VESSEL		P.V. RUPT. 1×10^{-8} OXIDIZING ATMOSPHERE	P.V. RUPT. 1×10^{-7} NONOXIDIZING ATMOSPHERE		
R PROBABILITIES	2×10^{-9}	2×10^{-8}	1×10^{-7}	1×10^{-8}	
SUMMATION OF ALL ACCIDENT SEQUENCE PER RELEASE CATEGORIES					
MEDIAN (50% VALUE)	1×10^{-6}	6×10^{-6}	2×10^{-5}	2×10^{-6}	1×10^{-4}
LOWER BOUND (5% VALUE)	1×10^{-7}	1×10^{-6}	2×10^{-6}	2×10^{-7}	1×10^{-5}
UPPER BOUND (95% VALUE)	8×10^{-6}	3×10^{-5}	8×10^{-5}	1×10^{-5}	1×10^{-3}

Source: NRC, 1975.

Note: The probabilities for each release category for each event tree and the sum for all accident sequences are the median values of the dominant accident sequences summed by Monte Carlo simulation plus a 10% contribution from the adjacent release category probability. External events not included. See table 5-3 for nomenclature.

TABLE 5-3. KEY TO BWR MARK I ACCIDENT SEQUENCE SYMBOLS

A	- Rupture of reactor coolant boundary with an equivalent diameter of greater than six inches.
B	- Failure of electric power to Engineered Safety Features.
C	- Failure of the reactor protection system.
D	- Failure of vapor suppression.
E	- Failure of emergency core cooling injection.
F	- Failure of emergency core cooling functionability.
G	- Failure of containment isolation to limit leakage to less than 100 volume percent per day.
H	- Failure of core spray recirculation system.
I	- Failure of low-pressure recirculation system.
J	- Failure of high-pressure service water system.
M	- Failure of safety/relief valves to open.
P	- Failure of safety/relief valves to reclose after opening.
Q	- Failure of normal feedwater system to provide core make-up water.
S_1	- Small pipe break with an equivalent diameter of about 2" to 6".
S_2	- Small pipe break with an equivalent diameter of about 1/2" to 2".
T	- Transient event.
U	- Failure of HPCI or RCIC to provide core make-up water.
V	- Failure of low-pressure ECCS to provide core make-up water.
W	- Failure to remove residual core heat.
α	- Containment failure due to steam explosion in vessel.
β	- Containment failure due to steam explosion in containment.
γ	- Containment failure due to overpressure - release through reactor building.
γ'	- Containment failure due to overpressure - release direct to atmosphere.
δ	- Containment isolation failure in drywell.
ε	- Containment isolation failure in wetwell.
ζ	- Containment leakage greater than 2400 volume percent per day.
η	- Reactor building isolation failure.
θ	- Standby gas treatment system failure.

Source: NRC, 1975.

TABLE 5-4. BWR MARK I CONTAINMENT FAILURE FREQUENCY BY TYPE OF FAILURE AND RELEASE CATEGORY

Type of Failure	Total Frequency	BWR Release Category			
		1	2	3	4
α	3.4×10^{-7}	3.1×10^{-7}	3.1×10^{-8}	3.1×10^{-9}	0
β	2.9×10^{-8}	2.4×10^{-9}	2.4×10^{-8}	2.4×10^{-9}	0
γ	2.5×10^{-5}	2.1×10^{-7}	2.1×10^{-6}	2.1×10^{-5}	2.1×10^{-6}
γ'	4.0×10^{-6}	3.3×10^{-7}	3.3×10^{-6}	3.3×10^{-7}	3.3×10^{-8}
δ	3.2×10^{-9}	0	0	0	3.2×10^{-9}

Source: Levy, 1983 (Table 4-7)

Note that 10% is placed in adjacent categories, and 1% is placed in the next adjacent category.

External events not included.

TABLE 5-5. DOMINANT ACCIDENT SEQUENCES IN THE BROWNS FERRY MARK I PLANT

Sequence	Frequency	Containment Failure Mode Frequency*		
		α	γ'	γ
$T_UR_BR_A$	9.7×10^{-5}	9.7×10^{-9}	1.9×10^{-5}	7.8×10^{-5}
T_UB	5.1×10^{-5}	5.1×10^{-9}	1.0×10^{-5}	4.1×10^{-5}
$T_PR_BR_A$	2.8×10^{-5}	2.8×10^{-9}	5.6×10^{-6}	2.2×10^{-5}
$T_KR_BR_A$	9.3×10^{-6}	9.3×10^{-8}	1.9×10^{-6}	7.4×10^{-6}
$T_UQR_BR_A$	4.1×10^{-6}	4.1×10^{-10}	8.2×10^{-7}	3.3×10^{-6}
T_ABM	3.7×10^{-6}	3.7×10^{-10}	7.4×10^{-7}	3.0×10^{-6}
$T_PKR_BR_A$	1.6×10^{-6}	1.6×10^{-8}	3.2×10^{-7}	1.3×10^{-6}
$T_PQR_BR_A$	1.2×10^{-6}	1.2×10^{-10}	2.4×10^{-7}	9.6×10^{-7}
	2.0×10^{-4}	1.3×10^{-7}	3.9×10^{-5}	1.7×10^{-4}

Source: Mays, 1982.

*α = In-vessel steam explosion (BWR 1 release category)
γ' = Direct release to atmosphere (BWR 2 release category)
γ = Release through reactor building (BWR 3 release category)

External events not included.

TABLE 5-6. NOMENCLATURE USED IN THE BROWNS FERRY PRA

B	= Failure of control rod drives
K	= Failure of relief valve to close
M	= Failure to trip the recirculation pumps
Q	= Failure of the reactor core isolation cooling (RCIC)
R_BR_A	= Failure of the decay heat removal system
T	= Transient event
T_A	= Transient with power conversion system available
T_P	= Transient due to loss of off-site power (LOSP)
T_U	= Transient with the power conversion system unavailable

Three other Browns Ferry dominant sequences ($T_PR_BR_A$, $T_PKR_BR_A$ and $T_PQR_BR_A$) involve loss of off-site power and subsequent failure of the RHR system. The effect of this is the same as that discussed immediately above. Several hours elapse before the loss of pumping capabiity occurs.

The last two dominant accident sequences in the Browns Ferry plant (T_UB and T_AMB) involve failure of the control rod drive system to insert enough rods to shut down the reactor. In one of these two sequences, the power conversion system (PCS) is assumed also to be disabled because of a transient event. This leads to eventual core dryout because the RHR cannot remove energy at a sufficiently large rate, and core dryout develops because of liquid boiloff. In the second instance, the PCS is available but the turbine by-pass valve cannot pass more than 30 percent of the rated steam flow. If the reactor recirculation pumps cannot be tripped, the reactor power may remain above the 30 percent level and gradual boiloff and core dryout would occur.

Tables 5-4 and 5-5 can be compared to illustrate the differences in the frequency of containment failure as revealed by two different Mark I PRA investigations. The IREP data in Table 5-5 indicate an overall frequency of containment failure (α, γ' or γ types) to be 2×10^{-4} events per reactor year. The RSS data in Table 5-4 (which apply to the Peach Bottom plant rather than the Browns Ferry plant used in the IREP) total 3×10^{-5} containment failure events per reactor year. Consequently, the estimated frequency of containment failure is approximately an order of magnitude higher in the IREP Mark I plant than in the RSS

Mark I plant according to the two different PRAs. In both instances though, overpressure-induced failure of the containment is the most likely cause of radioactive release. Recall, though, that in neither study are external events included.

5.3 BENEFITS FROM MITIGATION SYSTEM INSTALLATION IN THE MARK I

The decision to add severe accident mitigation features to Mark I plants can be guided by the expected benefit in doing so. One measure of benefit is the so-called 'value' of the candidate mitigation system. For the purposes of this discussion, value is defined to be the annual risk averted (as expressed in man-rem, acute fatalities or latent fatalities) through the mitigation of consequences. Section 2.2 discussed how the quantity of risk averted (ΔR^i) could be calculated knowing the expected frequency of containment failures and the radioactive release associated with each failure. Tables 5-4 and 5-5 provide a breakdown of the Mark I containment failure frequencies by mode of failure for two different Mark I plants. Mitigation systems will normally be designed so that one or more of the specific containment failure modes are precluded. The accident consequence in terms of man-rem and acute fatalities for each of the potential BWR release categories is shown in Table 4-6. Each containment failure mode as shown in Tables 5-4 and 5-5 is characterized by one or more of these release categories depending on the accident sequence.

Other investigators have also considered the benefits of risk reduction through the installation of mitigation systems in the Mark I plant. For example, the use of filtered venting from the containment following a severe accident and the use of an ex-vessel core retention device are being studied at the SNL (Fish, 1983; Benjamin, 1981). Similarly, EPRI has considered the introduction of a filtered vent on all BWR pressure-suppression-type containments as well as on the ice condenser containments (Levy, 1981). These studies quantified risk reduction in terms of man-rem, acute fatalities, latent cancers, and/or property damage averted. Generally, only internal events were considered and monetary values were not assigned to the results. The studies provide a detailed discussion of the practical considerations involved when introducing mitigation devices in the Mark I plant.

The determination of potential benefits in terms of risk averted for a Mark I plant will be site- and plant-specific

and will depend on the accident sequence frequency leading to core melt, the probability of containment failure (which depends on the characteristics of the individual plant Engineered Safety Features), and the consequences (which are site-specific). This information, to the extent to which it can be drawn from PRA results for the Peach Bottom and Browns Ferry plants, will be used below to quantify risk reduction benefits.

Table 5-7 shows the risk presented annually by the Peach Bottom and Browns Ferry plants in terms of man-rem and acute fatalities per year. This information can be used directly to compute the benefits of selected generic mitigation systems. In order to quantify the risk reduction potential of various mitigation systems applicable to the Mark I, the risk reduction ratio is computed. Values applicable to the Mark I plants are shown in Table 5-8.

TABLE 5-7. CONTRIBUTION TO RISK BY EACH FAILURE MODE

Failure Mode	Man-Rem Per Year	
	Peach Bottom	Browns Ferry
α	0.74	0.29
β	0.051	---
γ	24	151
γ'	7.0	70
δ	0.0013	---
All	32	221.3

Failure Mode	Acute Fatalities Per Year	
	Peach Bottom	Browns Ferry
α	5.8×10^{-7}	2.2×10^{-7}
β	14×10^{-7}	---
γ	7.5×10^{-5}	5.1×10^{-4}
γ'	19×10^{-5}	19×10^{-4}
δ	1.2×10^{-8}	---
All	27×10^{-5}	24×10^{-4}

Basis: RSS source term and methodology.

TABLE 5-8. RISK REDUCTION FACTORS FOR MARK I SEVERE ACCIDENT CONSEQUENCE MITIGATION SYSTEMS

Containment Failure Mode Eliminated by Mitigation System	Based on Acute Fatalities		Based on Man-Rem	
	RSS*	IREP+	RSS*	IREP+
α	1.02	1.0	1.0	1.0
β	1.0	---	1.0	---
γ	4.0	3.2	3.6	4.7
γ′	1.3	1.46	1.42	1.26
δ	1.0	---	1.0	---
Total	32.0	736	54	104

*Peach Bottom
+Browns Ferry

External events not included.

In Section 5.4 specific mitigation systems will be suggested. Venting systems and/or containment heat removal systems will probably be needed to prevent overpressure failure. Note also that the full risk reduction benefit may not be realized if only this particular type of failure is averted. Subsequently, other containment integrity threats may develop during the course of a single severe accident. For example, the loss of containment integrity may occur due to basemat penetration by the hot core debris well after overpressure failure would have occurred without a vent or heat removal system.

5.4 MITIGATION OPPORTUNITIES FOR MARK I CONTAINMENTS

Section 5.3 showed that the risks associated with Mark I BWR plant operations are primarily due to overpressurization events following core-melt incidents. Furthermore, the risks associated with these events are sufficiently large that they encourage the search for mitigation devices capable of averting the associated risk. In this section a series of steps will be taken to identify the generic mitigation systems capable--in principle--of achieving risk reduction in Mark I plants. The selection will be guided by this type of threat imposed on the Mark I containment by severe accidents, by the accident environmental conditions,

and by the possible interaction with other plant systems. General design criteria will be suggested.

Finally, candidate mitigation systems will be listed that meet the functional requirements deemed appropriate for risk reduction. In general, the candidate systems constituting a mitigation system package must encompass a range of severe accident consequences in addition to the overpressurization that would otherwise first lead to containment failure and subsequent release of radioactivity. The cumulative listing of the severe accident conditions that form an envelope of constraints on the mitigation systems is given in this section.

5.4.1 Cumulative List of Severe Accident Conditions in the Containment

A core-melt accident event in a Mark I system would be characterized by a range of containment and other plant conditions depending on the specific accident sequence. Because the mitigation system must operate successfully in the presence of any of these conditions, they are all assumed to exist simultaneously. The cumulative list of accident conditions defines a major part of the operational environment for the mitigation systems. Relevant conditions are as follows:

1. Station blackout exists and power cannot be restored (controls and instruments cannot be relied upon or are inoperative).

2. Containment atmospheric heat removal systems (sprays) are inactive.

3. Suppression pool heat removal systems are inactive.

4. The suppression pool is saturated at a pressure corresponding to the containment design pressure.

5. Degradation of the core has begun. It will eventually melt through the vessel and drop or be forced out (partially or completely) along with a large portion of the RPV's lower steel internals.

6. A sizeable portion of the hot Zircaloy fuel cladding has reacted with steam to form hydrogen gas. This gas is released to the containment in quantities as large as several thousand pounds.

7. Plant operators are unavailable or otherwise unable to initiate accident management functions

that may be possible despite station blackout and the loss of emergency heat removal systems.

8. The ultimate heat sink is available but cannot be accessed by normal and emergency fluid flow circuits.

One other condition may be present if the accident sequence is of the ATWS type. In such a situation, the dryout and subsequent degradation of the core is preceded by very high steaming rates which must be handled by the safety/relief valves. This condition, as well as the others listed above, is imposed on a Mark I primary containment that is inerted with nitrogen at the initiation of the severe accident.

5.4.2 Criteria for Selecting Mark I Mitigation Systems

Several mitigation system design criteria are considered along with the envelope of cumulative and specific accident conditions. This combination then serves to establish the conceptual nature of the mitigation systems. The design criteria are developed on the basis that the mitigation systems truly represent a last bulwark to potentially major radioactive releases from the plant. As such, they must operate with the highest possible level of reliability despite a harsh accident environment. The following criteria are intended to assure a benign ultimate plant condition:

1. Mitigation systems must be designed to circumvent or fully encompass the range in uncertainty associated with severe accident phenomena.

2. Mitigation systems must be passive in their operation to the maximum extent possible. No reliance should be placed on the plant's normal or emergency electrical basis. If energy sources are required, they should be redundant and dedicated to the mitigation system.

3. The mitigation systems must not compromise the operation of normal plant equipment nor the plant's Engineered Safety Features.

4. Mitigation systems must be designed to withstand the most severe environment under which they must operate.

5. Controlled venting (i.e., filtered or scrubbed venting) are preferred to the uncontrolled release of containment contents through a ruptured primary structure. Venting, if needed,

should take place automatically and vent paths should automatically reclose when pressure levels fall.

6. Operator actuation of the mitigation systems should not be required.

7. The path of core debris exiting the reactor vessel must be controlled in order to direct the material to a designated final resting place.

5.4.3 Potential Severe Accident Conditions to be Mitigated

Subsection 5.4.1 identified the primary or global conditions that characterize the main severe accident conditions. These conditions result in certain threats to Mark I primary containment integrity. It is these threats in turn which give rise to the overpressurization of the containment as highlighted in Section 5.3. Specific conditions assumed to occur in the absence of proper accident mitigation systems include the following:

1. Thermally-induced decomposition of concrete in the pedestal region by ex-vessel core debris. Concrete decomposition results in copious quantities of noncondensible gases.

2. Rapid containment pressure rise due to steam generation by the sensible heat released from the ex-vessel core debris as it contacts water. The core debris may drop into water spilled earlier either in the pedestal or into the suppression pool.

3. Ex-vessel combustible gas production. The reduction of water vapor and carbon dioxide (both are generated in heated concrete) by hot steel or Zircaloy in the core debris produces hydrogen and carbon monoxide. Hydrogen produced in this phase would add to any generated in the RPV and adds to the flammability risk.

4. Continuous production of decay heat. Thermal energy will be generated continuously by fission products in the core debris long after core degradation begins.

5. Rapid steam production by an ATWS event.

Several specific accident conditions are not included in this list because PRA results indicate they present minimal

threat to the Mark I containment. Conditions of this type include in-vessel and ex-vessel steam explosions and--as a consequence--missiles.

5.4.4 Cumulative List of Mitigation Functions to be Performed

A series of mitigation functions suggest themselves following consideration of the conditions identified in the preceding three subsections. These are generic functions and must--prior to implementation--be made specific with respect to capacity, placement, and equipment type. The primary thrust of the functions is to alleviate the overpressurization threat to the containment as well as any subsequent threats. Four important functions appear to suffice:

1. Large capacity primary containment venting to accommodate ATWS-induced steam generation in order to avoid overpressurization. Vent paths should reclose automatically when the threat of overpressurization by this "clean" steam is removed. This same vent path or a parallel path may be used later for the release of smaller quantities of radioactive steam produced during and/or following core degradation. Filtering of radioactive steam is required to minimize hazardous releases. Vacuum breakers must be used to admit outside air so as to prevent containment collapse. The steam being vented carries noncondensible gases with it so when the containment cools, the condensing steam creates a negative pressure condition.

2. Long-term containment heat removal. The heat removal system must transfer decay, sensible, latent and chemical-reaction-generated heat from the containment to the ultimate heat sink.

3. Control and confinement of the path of ex-vessel core debris with ultimate deposition in a localized region. The final core debris resting place must be thermally connected to the containment cooling system for heat removal and thermally isolated from concrete structures.

4. Thermal protection of containment penetrations. The sealants, packings, and potting materials used with penetrations may have a temperature rating of 250° to 350°F.

5.4.5 Candidate Mitigation Systems for Mark I Containments

Mitigation systems appropriate for use in Mark I containments have been discussed in Hammond (1982), Swanson (1983), and Ahmad (1983), and are reviewed in a companion report in this series for Task 2. The mitigation methods considered include filtered vent systems, containment heat removal, and core retention systems.

A number of different filtered vent options of general applicability have been proposed (Levy, 1981; Ahmad, 1983; Reilly, 1982). Designs suitable for use in a Mark I containment have been examined in Levy (1981) and Harper (1982). Calculations for the case of a release of 5000 lb of hydrogen over 30 min indicate that a vent rate of 27 lb/s (12,000 cfm at 97 psia) should be sufficient in a system derived from the existing SGTS (Levy, 1981). Other designs (Benjamin, 1981) will accommodate gas flows as large as 250,000 cfm with various filtering arrangements. Even larger capacity vents for ATWS-induced steam generation can be considered.

Hydrogen control is provided in the Mark I by preinerting the containment with nitrogen. Although additional measures for hydrogen control might be desirable to reduce the danger of overpressurization and the flammability problems that can arise during deinertion, it is not clear at this point that such measures are either needed or justified.

Heat pipes, large single condenser systems, or water sprays might be used to provide containment heat removal. Heat pipe systems have been studied in detail for this purpose (Ahmad, 1983). Two core retention systems have also been suggested that could be retrofitted into existing reactors. These are the flooded pebble bed system (Swanson, 1983) and the water-cooled crucible installed in a tunnel below the pedestal (Hammond, 1982). In new construction, there are a number of other core retention systems that may be considered. All the systems mentioned above appear feasible in the Mark I containment. Qualifications concerning their use are discussed in greater detail in the accompanying Task 2 report.

A variety of insulation and radiation barriers could be adapted to the containment penetrations for thermal protection. Water sprays, if directed at penetrations, may also afford adequate protection in the event hot core debris is released to the containment.

5.5 SUMMARY

The Mark I containment is the first standardized containment built for GE BWR nuclear plants. It has since been superseded by the more advanced Mark II and III configurations. Twenty-four U.S. plants are in this category with the oldest active plant--Oyster Creek I in New Jersey--having received its operating license from the NRC in April 1969. Plant sizes range from 53 to 1067 MWe.

The Mark I containments are relatively small compared to PWR plants with similar name plate ratings. A net internal free volume of 270,000 ft^3 is common. Because of the small volume, the primary containment is inerted with nitrogen as a form of hydrogen combustion/detonation control. The suppression pool is relied upon to condense steam released from a primary system LOCA. The passive nature of the suppression pool enhances its reliability in the event of a LOCA.

The types of Engineered Safety Features included in Mark I plants to accommodate DBAs are similar to those used with Mark II and III plants. The safety features of primary importance with respect to accidents involving core melt are the primary containment barrier, the containment isolation system, the SGTS, the pressure suppression pool, the containment and pool cooling system, the ECCS, and the ultimate heat sink. Proper functioning of these systems is normally relied upon to prevent the occurrence of Class 9 accidents (i.e., core-melt accidents).

Mark I primary containments (with the exception of the two Brunswick units) are built of steel plate, with the surrounding reactor building functioning as a secondary containment and external missile shield. The primary containment is in the form of an inverted light bulb with the torus-shaped suppression pool structure located underneath and connected to the light bulb portion with large vent pipes.

Information pertaining to the expected frequency of severe accidents in the Mark I plant and their likely consequences is drawn from two sources. The RSS applied PRA techniques to the Peach Bottom plant, and the IREP employed a similar exercise for the Browns Ferry Unit 1. However, in the latter case, reference was made to the RSS for severe accident consequence information. For the Peach Bottom plant, it was concluded that the median value for the frequency of core-melt accidents was 3×10^{-5} events per reactor year.

The severe accidents of greatest probability are those involving containment overpressure failure. These sequences are precipitated by a transient event followed by failure of the RHR system or other safety systems. These sequences are identified as TW, TC, and TQUV. The radioactive release associated with these sequences is identified by the RSS terminology as BWR 3. A value of 0.89×10^6 man-rem is assumed to be the release consequence (as it was in the RSS). Severe accidents due to LOCA events of all sizes are less likely to occur in the Peach Bottom plant.

A somewhat different core-melt frequency was found in the Browns Ferry PRA work done as part of the IREP work. It was found there that the aggregate frequency of all core-melt incidents totals 2×10^{-4} events per reactor year--nearly ten times as large as that characterizing the Peach Bottom plant according to the RSS. As with the Peach Bottom plant though, the dominant contribution to core melt in the Browns Ferry plant is from transient events followed by failure of the RHR system. These results suggest that the type of mitigation systems proposed for one of these plants would serve well in the other. Recall, though, that external events were not considered in either the RSS or IREP studies.

ATWS-type events are one of the dominant accident sequences in both the Mark I plants discussed above. While not the most probable, they do have the greatest consequence when they occur because of the rapidity with which the containment is overpressurized. The influence of the 3A-Fix for ATWS events is not included in the material referenced with respect to Peach Bottom and Browns Ferry.

Other work in addition to the suggested ATWS-3A-Fix for Mark I plants has a potential impact on the risk and consequently the value of mitigation systems. A detailed study of the Mark I plant structure by Battelle Columbus Laboratory (1977) raises the possibility that containment failure may not take place at locations assumed in the RSS. In particular, failure in the drywell rather than the wetwell would significantly reduce the amount of scrubbing associated with radioactive releases following core melt. Therefore, the accident consequence would rise accordingly.

The uncertainty regarding the location of the Mark I containment failure is not the only factor which renders risk projections speculative to a degree. The possible influence of external events is not included in the PRA work to date for Mark I plants. Furthermore, considerable effort is now being applied to improve plant operator effectiveness. All these factors act in concert to weaken confidence in the

numerical values highlighted above. Since no other more complete information is available at this time, the Mark I PRA results for Peach Bottom and Browns Ferry are used without modification.

Mitigation systems proposed for Mark I plants are selected with the assumption that they must perform their intended functions in spite of whatever equipment and operator deficiencies characterize any of the dominant core-melt accidents. This design basis translates as requiring operation under the conditions of (1) station blackout, (2) operator unavailability, (3) saturated suppression pool, (4) extensive metal-water reaction, (5) release of molten core materials from the vessel, (6) ATWS steaming rates through safety/relief valves, (7) rapid steam generation from core debris falling into water, and (8) decay heat generation. Candidate mitigation systems are those able to maintain primary containment integrity despite these events.

The functional requirements of the candidate mitigation systems are the ability to (1) vent large amounts of steam produced by an ATWS event prior to core disruption, (2) remove residual heat from core debris and the containment, (3) control the pathway followed and the ultimate location of core debris within the containment, and (4) protect containment penetrations against high-temperature-induced transients.

A number of mitigation systems are attractive candidates. These include large capacity/self-closing steam vents, low capacity/filtered vents, containment sprays (for containment heat removal), core retention in a crucible, and hydrogen recombiners or burners. These systems would have to operate with dedicated power sources. The selection of systems to be used would be plant-dependent and would involve considerations of cost, compatibility with other plant systems, dependability, durability and ease of installation. Some of them could serve to reduce the likelihood of severe accidents as well. For example, a new, dedicated RHR system would lower core-melt frequency by serving as a backup in the case where the RHR systems classified as an Engineered Safety Feature fail to function.

The value of severe accident consequence mitigation can be estimated as a function of the type of containment failure and associated radioactive release averted. The averted risk varies also between the two Mark I plants for which detailed accident frequency and consequence data are available. The risk averted due to prevention of overpressure failures of the containment has the largest value for both plants, roughly 220 man-rem per year for Browns Ferry and 32

man-rem per year for Peach Bottom. This translates into risk reduction factors greater than 100 and 50 respectively. Averting all the other failures highlighted by the PRAs has lesser values. Of course, some of these other failure modes such as basemat melt-through become more important and must also be mitigated if the full value of averting overpressure failure is to be realized.

CHAPTER 6. PWR ICE CONDENSER CONTAINMENT

The historical increase in PWR power level and associated coolant energy has caused attention to focus on alternative containment concepts. Traditionally, dry containments relied on a combination of structural strength and volume to contain potential steam and radioactivity releases from the reactor primary system in the event of an accident. The reinforced-concrete containment (as opposed to a heavy wall steel containment requiring field stress relief) was one response to the need for larger yet economical containments. Another approach to containment design was initiated by Westinghouse in 1965 (Weems, 1970). This approach provides a large static heat sink (i.e., ice) inside the containment to absorb accidental energy release from the reactor's primary system.

Appendix D indicates that ten U.S. nuclear plants have ice condenser containments. Approximately half of the units have been granted operating licenses to date. The oldest unit, Cook 1 at Benton Harbor, Michigan, received its operating license in October 1974. All the units have large name-plate electrical power ratings, ranging from 1054 MWe to 1180 MWe. It is precisely the large capacity of the units which spurred interest in the ice condenser type of containment. Appendix D also points out that eight of the ten ice condenser plants will have secondary containments. Figure 6-1 is a sectional view of a plant in this containment category.

The ice condenser containment consists of three major compartments. As shown in Figure 6-1, the reactor and primary piping are located in the lower compartment. This is sometimes referred to as the upstream compartment. A second compartment is the upper (or downstream) compartment. It acts as a receiver to contain air driven from the lower compartment by steam released from a break in the reactor primary system. The operating deck separates the upper and lower compartments and serves as a low leakage barrier and missile shield (the enclosure around the steam generators and pressurizer form continuations of this barier). The third, or transfer, compartment contains ice. This compartment serves as a large heat exchanger. Steam is condensed within it, and air plus other noncondensible gases are conveyed by it to the upper compartment during an accident. The ice condenser, or middle compartment, is essentially a cold storage room holding approximately 2,300,000 lb of borated ice in perforated metal baskets. Development of the ice condenser concept required a thorough study of ice

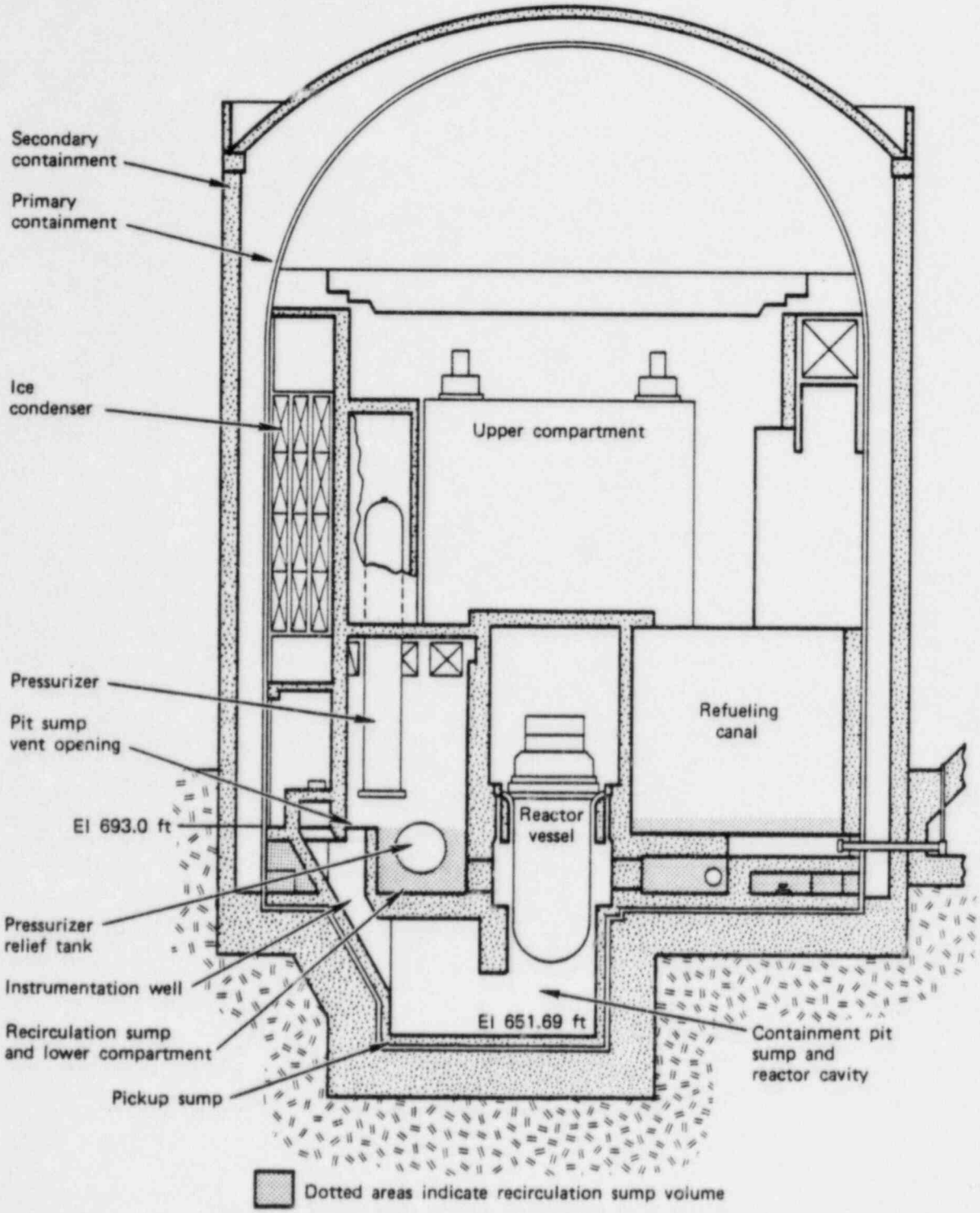


Figure 6-1. Ice condenser containment.

densification, creep, sublimation, and fallout under seismic loading. Small and large-scale functional tests of the system have been performed (Dragoumis, 1968). The system's steam condensation capabilities have never been required in an operating plant.

The ice condenser containment is discussed in further detail in Section 6.1. The containment is an Engineered Safety Feature of the plant and, because of this, is designed to more stringent standards than other plant features. The DBA set provides an envelope of severe accident environments that the Engineering Safety Features must accommodate without adverse radiation release from the plant (see Appendix B). Other Engineering Safety Features directly complement and make possible the successful operation of the containment. Complementary Engineering Safety Features include the containment cooling system, atmospheric control, isolation features, and the secondary containment. These systems are also discussed in Section 6.1, and reference is made to the DBA conditions they can accommodate plus other conditions they cannot.

PRA studies have highlighted accident sequences beyond the design basis that can cause containment failure (containment failure modes are discussed in Appendix A). The dominant accident sequences leading to failure are identified in Section 6.2. Mitigation systems are then proposed in Section 6.3 and 6.4 that may be able to alleviate--at least to some extent--the consequences of severe accidents (i.e., Class 9 accidents). The degree to which this is theoretically possible is estimated below along with the value of any risk reduction. The calculation of quantitative change in risk is made through modifications of the PRA results. The PRA study results are available for the Sequoyah plant. This unit is treated as a surrogate for all the ice condenser plants. It is assumed in Section 6.3 that certain accident-driven radioactivity release categories can be wholly eliminated through the action of selected ideal mitigation systems (later program tasks will estimate the actual extent to which real-world mitigation systems can match this ideal performance).

Based on the value of the various mitigation systems, recommendations are made for further detailed study of mitigations systems in future tasks (Section 6.5).

6.1 DESCRIPTION OF THE ICE CONDENSER CONTAINMENT AND RELATED SYSTEMS

This section of Chapter 6 provides a description of the ice condenser plant environment into which severe accident

mitigation systems may be fitted. The emphasis in the discussion is on the nuclear reactor itself, the reactor internals, the ECCSs, the primary and secondary containment, and the Engineered Safety Features associated with the primary containment function. Items in the last category include the containment isolation system, the long-term containment heat removal system, and the combustible gas control equipment.

The reactor description highlights the materials potentially involved in a core meltdown. Of interest are the quantity of fuel, cladding, and steel available as meltdown constituents (the mixture is referred to as corium). The lower head, lower head contents, and lower head penetrations are relevant to the release of core materials from the vessel. The ECCSs are briefly described. However, in the balance of the mitigation system analysis it is assumed the ECCS has failed to function for an unspecified reason or reasons.

A severe accident mitigation system is likely to depend--at least in part--on a fully functioning primary containment. Therefore, the containment dimensions, design pressure, construction materials, internal heat sinks, and leakage rate are highlighted in the discussion. The complementary systems needed to maintain containment isolation, heat removal and combustible gas control are also described. The existing plant Engineering Safety Features that are designed to accomplish these same functions under DBA conditions are included in the discussion of this section. Reference is made to Appendices A and B for a review of the plant safety system strategy and DBA set.

6.1.1 Reactor and Primary System

The nuclear reactor system used exclusively in ice condenser plants is the four-loop Westinghouse PWR (see Figure 6-2). The thermal output of each of the ten U.S. plants in this category is approximately 3425 MWt. The reactor core design is similar to other Westinghouse models of comparable vintage and is described in Section 7.1. Likewise, the Engineered Safety Features associated with the reactor's primary system are analogous to those of Section 7.1.

The PWR fuel weight is approximately 220,000 lb, and the cladding weight is approximately 51,000 lb in an ice condenser plant. Nominal primary system operating pressure is 2250 psia, and the total coolant flowrate is about 140,000,000 lb/hr. The core diameter is about 133 inches, and the height is about 144 inches. The in-core instrument tubes enter the reactor vessel from beneath. Consequently, the reactor cavity contains a keyway through which the

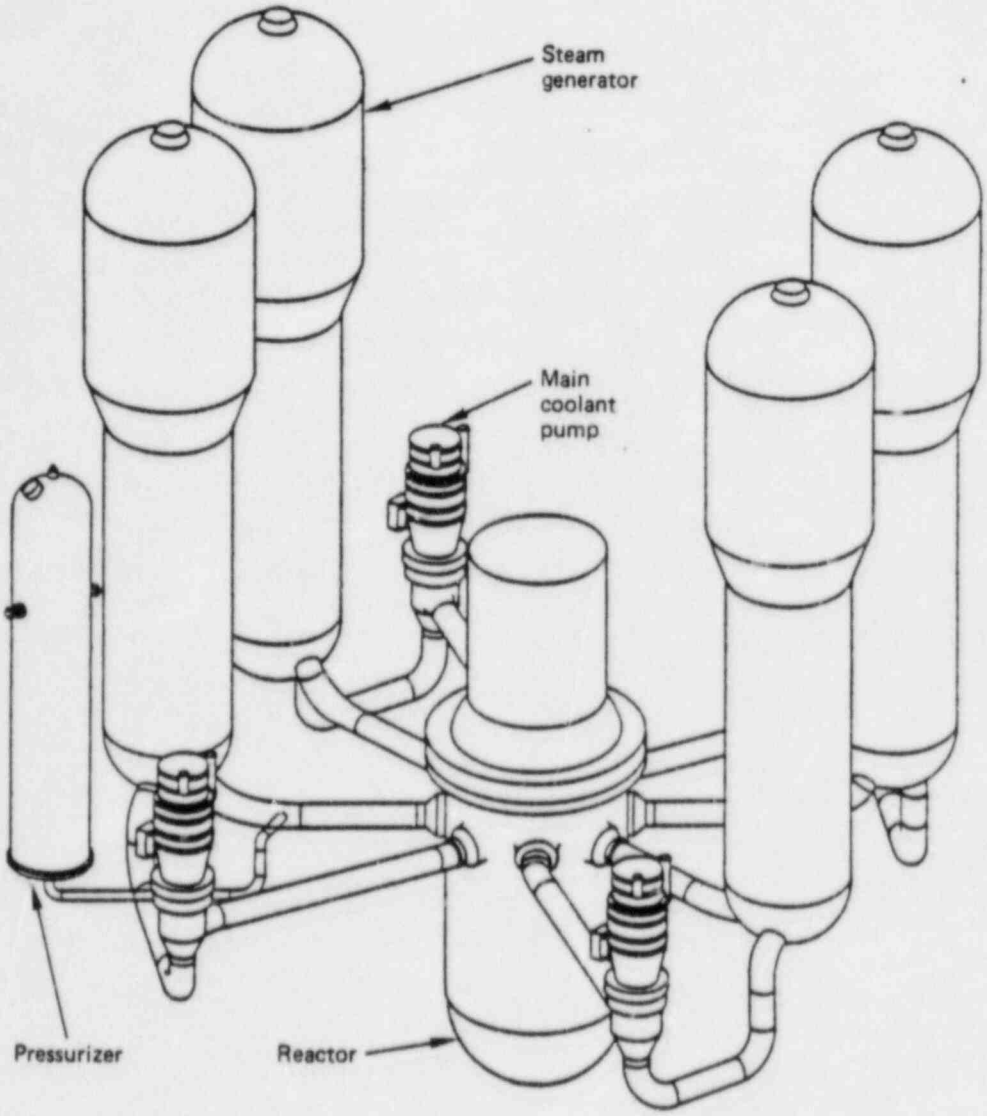


Figure 6-2. Simplified diagram of four-loop nuclear steam supply system.
Source: Spencer 1982

instrument tubes are routed away from the vessel and upward to a seal table on the operating floor of the containment. The containment basemat and keyway under the reactor vessel are shown in Figure 6-1.

6.1.2 Primary Containment and Ice Condenser

Two types of construction have been used with ice condenser containments. The first employs a deformed-bar reinforced-concrete vertical cylinder, a hemispherical dome, a flat base and a steel liner for the primary containment. The concrete wall thickness is about 3 ft. The volume of each of the ten structures listed in Appendix E is roughly 1,200,000 ft^3, and each has a diameter of about 115 ft and a height of 155 ft. The Cook 1 and 2 plants were constructed in this manner. The other ice condenser plants use a free-standing steel cylinder, a hemispherical steel dome, and a flat concrete base with a steel membrane liner. The operating deck that divides the upper and lower containment compartments is built of reinforced concrete in all cases.

The design pressure of the containment is 12 to 15 psig. The negative design pressure is about 0.5 psig, and vacuum breakers may be present in the containment wall to prevent this value from being exceeded. The containment is also designed to accommodate snow, wind, external missiles, and earthquake loads. With the exception of the Cook 1 and 2 plants that lack a secondary containment, leakage through the primary containment is to the secondary containment. The design leakage rate is 0.2 to 0.5 percent of the containment volume per day at the design pressure. Leak tests are conducted periodically at 12 to 16 psig.

The ice condenser chamber forms a passage between the upper and lower compartments. It holds approximately 2,300,000 lb of borated ice in perforated metal baskets. The ice is in flake form less than 1/4 inches thick. Ice is conveyed into the baskets pneumatically from an off-site ice-making machine. The ice baskets are stacked so that channels through and around them are created for the flow of steam and air. The walls and floor of the ice bed consist of insulated ducts carrying coolant from an air-refrigeration unit. The ice bed is maintained at a temperature of about -9.5°C.

In plan view the ice condenser forms a partial annulus, extending 300° around the perimeter of the primary containment. It is approximately 95 ft high and 13 ft wide. Insulated spring-loaded inlet and outlet doors are located at the lower and upper ends respectively of the ice bed.

The operating principle of the ice condenser is straightforward. If a coolant pipe breaks in the lower compartment, steam and water are released to the compartment. The increase in lower compartment pressure pushes steam and air through the lower ice condenser inlet doors, through the ice condenser and through the outlet doors into the upper compartment. A pressure differential of <1 psi will open the doors. The steam partial pressure is rapidly reduced since nearly all of the steam is condensed. The subsequent peak containment pressure is normally set by the compression process occurring in the upper compartment which acts as a receiver. For the most severe DBA LOCA condition, the maximum upper compartment pressure will be approximately 10 psig. The pressure is expected to drop below 2 psig within several minutes (Dragoumis, 1968).

The ice condenser containment has several advantages over a conventional dry containment. Some of these are listed below.

1. The containment height and diameter are respectively 60 and 25 ft less than the dimensions of a comparable dry containment. Similarly, the design pressure is reduced 75 percent.

2. The lower pressures experienced during DBAs cause less leakage through the containment. The time the containment is at an elevated pressure during an accident is reduced also, thereby further reducing leakage of radioactive materials.

3. The thin-walled (less than 1-1/2 inch), free-standing steel containments do not require field stress relieving.

4. The condensate formed by melting ice provides a scrubbing action during steam passage.

5. Elastomeric liners rather than steel membrane liners may be used because of the low design pressure.

6. A large heat sink is always present in the containment. The latent heat of the 2,300,000 lb of ice is approximately 330 MWh.

7. The design is relatively insensitive to large variations in the important accident parameters such as the LOCA blowdown rate.

Several aspects of the ice condenser containment should be noted. The relatively small volume raises concern regarding possible hydrogen concentrations formed following metal/water reactions. To contend with this situation, glow plugs have been positioned near the exit doors in the Sequoyah plant. These are intended to cause burning of the hydrogen continuously rather than allow an accumulation of the gas and a possible subsequent detonation. Another feature of the containment is the reactor cavity in the containment basemat. This region is normally dry while the space below the reactor is filled to a large extent by the multiple instrument tubes (diameter about one inch) exiting from the bottom of the vessel.

The ice condenser containment also has certain drawbacks. Most notable is the common mode failure resulting if the melted ice and containment spray water do not reach the lower cavity sump. This is discussed in subsequent sections dealing with dominant accident sequences. Another is the effect of reflood loading on the upper head injection system.

6.1.3 Secondary Containment

Eight of the ten ice condenser plants listed in Appendix D have concrete secondary containments. Only the Cook 1 and 2 plants do not have this structure as a complement to the primary containment. In many cases, the secondary containment is referred to as the shield building. It shields the primary containment from external events such as tornado-induced missiles and provides biological shielding following DBAs. The Cook 1 and 2 plants have concrete primary containments. These thick concrete structures are adequate, in themselves, for missile and biological shielding. A steel primary containment does not provide adequate protection against missiles or have adequate shielding characteristics.

While the primary containment is the main leakage barrier in an ice condenser plant, the secondary containment plays an important role in controlling radioactive releases from the nuclear plant in the event of a severe accident. An annular space several feet wide (about 5 ft) exists between the two containment structures. This space is maintained at a slightly negative pressure (about -0.5 inch water column) following a LOCA. The annulus ventilation system (AVS) withdraws air from the annulus space and filters it prior to its release to the outside environment via the plant stack. Two full capacity AVSs are present. The AVS action is intended to prevent a pressure buildup in the annulus due to primary containment leakage or due to heat transfer from the primary containment. Filtering of the gases that are

released helps assure that the site boundary radiation levels are within statutory limits during DBAs and other abnormal occurrences. Gases are routed through a moisture separator, a particulate filter, a HEPA filter, and a fire-resistant charcoal filter. Because of its importance, the AVS is designed to be an Engineered Safety Feature. It is electrically operated and is connected to the plant's emergency buses.

The secondary containment is sometimes referred to as the shield or reactor building. Its cylindrical, reinforced-concrete walls are about 3 ft thick. It has a hemispherical concrete dome, and its flat concrete base is shared with the primary containment.

6.1.4 Isolation Devices

The containment isolation devices and strategy for their use in an ice condenser plant are essentially the same as for the dry containment type plant. Features of the isolation system in this latter category are discussed in Section 7.1. The presence of the ice condenser with the primary containment does not alter the need for, nor operational procedures associated with, containment isolation.

Containment isolation is dependent on the proper functioning of numerous seals and gaskets around containment penetrations which cannot be welded to the containment itself or to the containment liner. A variety of elastomeric materials are used for this sealing function. The materials, however, are susceptible to temperature, humidity, and radiation. Consequently, their performance is checked periodically when the overall containment leakage rate is measured while the containment is deliberately pressurized. The use of double seals in some instances allows individual units to be tested for leak-tightness. The containment temperature limitations are a consequence of the temperature sensitivity of elastomeric materials.

6.1.5 Long-Term Containment Heat Removal

Energy may be released to the lower compartment over a long time period in the event of a DBA or other accident. For example, even following a reactor scram action, the reactor is capable of producing energy at an initial rate of 6 to 8 percent initially, and less later of its full thermal output because of fission-product decay heating. The ice condenser long-term cooling system can remove decay heat plus other energy present from metal/water reactions and sensible heat sources in the primary reactor system. A water spray system in the upper containment compartment provides additional

cooling (the lower compartment has no cooling provisions other than those associated with the reactor itself). The spray cooling system is an Engineered Safety Feature and requires electrical power for actuation and operation. It is sized to remove all the decay heat following the time when essentially all the ice has melted (about 2 hr after scram in a large LOCA).

A simplified process diagram for the containment spray system is shown in Figure 6-3. Its major elements are the spray headers (two or four) in the upper compartment, the RHR heat exchanges (two or four), the circulating pumps (motor-driven, two or four), the containment pump, and the actuation system. Spray headers are made of heavy steel pipe and are suspended from the containment.

Two independent full sized spray systems are used (Liparulo, 1976). The flow rate of each is about 2600 to 3400 gal/min. Water can be drawn by the circulating pumps from either the refueling pool or the containment sump. Pump water is provided by melting ice, steam condensation, water released from a break in the reactor system, and/or return spray water. Small drain holes in the operating deck allow water to flow downward from the upper compartment into the sump. The small drain holes are the source of one of the big contributors to risk. The water sprays moderate both temperature and pressure in the containment and provide a radioactivity scrubbing action.

Actuation of the containment spray system is automatic. A high containment pressure (about 3 psig required) or a high containment radiation level can initiate the spray circulation pumps.

Heat removal from the containment spray water takes place in the RHR heat exchangers. Each of these units has a rating of approximately 90,000,000 Btu/hr (25 MW). Cooling water is supplied to the exchangers from the plant's ultimate heat sink. Electric power is necessary to operate the secondary water flow system. Redundancy and stringent design standards apply to the secondary system since it also is an Engineering Safety Feature.

A small atmospheric recirculation system within the containment is another Engineering Safety Feature. Two small fans return air from the upper to the lower compartment following a LOCA blowdown. These fans help assure an even distribution of air throughout the containment and assist in moving air through the ice condenser (Weems, 1970). The containment cooling system used during normal plant operations is not an Engineering Safety Feature. These units normally

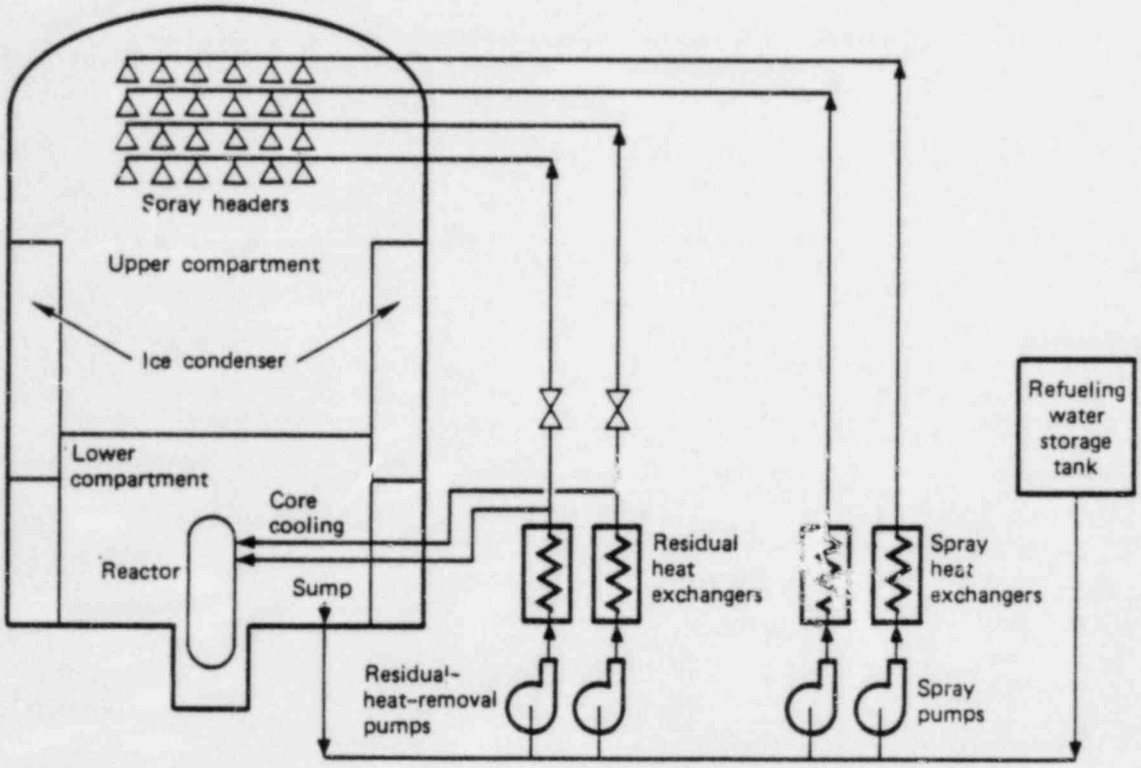


Figure 6-3. PWR ice-condenser long-term containment heat removal system.

hold the atmospheric temperature in the 60° to 120°F range under nonaccident conditions.

6.1.6 Combustible Gas Control

Electric hydrogen recombiners provide hydrogen removal capacity within the condenser plant's primary containment. Two full capacity, redundant systems are provided, and each has the capacity necessary to limit the hydrogen concentration to 4 percent or less following a DBA. The recombiners are Engineered Safety Features and are designed to withstand all normal loads as well as accident loads. They are connected to emergency buses for electrical power in the event that the off-site power source is lost.

The hydrogen recombiner consists of a thermally insulated metal duct with electric resistance heaters located inside. The heaters raise the temperature of a continuous flow of gas from the containment. Air is drawn into the unit by natural circulation forces. It is heated to 1150° to 1400°F at which point the recombination reaction between hydrogen and oxygen takes place. The recombiner units are positioned so that they process containment air containing hydrogen at a concentration which is generally typical of the concentration throughout the containment. Electric fans are positioned in some of the containment's dead end compartments in order to prevent the accumulation of locally high hydrogen concentration levels.

6.2 DOMINANT FAILURE MODES OF ICE CONDENSER CONTAINMENTS

Ice condenser containments have not received as much attention as the large dry PWR containments primarily because there are many fewer of them. While much different, the analyses of dry containments should be applicable to ice condenser containments if one accounts for the decreased volume and reduced design pressure. The large dry containment is addressed in Section 7.2 and will not be repeated here. Results of recent work addressing more specifically the ice condenser containment (Carlson et al., 1981; Levy et al., 1981; Reilly et al., 1982) will be summarized, and the dominant accident sequences and containment failure modes will be presented.

Calculations of containment response during core-melt accidents were performed by SNL (Carlson et al., 1981) as part of the RSSMAP. The calculations are stated to have been done on a best-estimate basis. Systematic use of RSS-based techniques coupled with results from specific RSS analyses permitted SNL to identify, prior to more detailed analysis, those systems not contributing to risk. It was therefore

argued that an exhaustive search and evaluation of a large number of systems in many accident scenarios could be avoided. The analysis started with only those systems that were thought to make a significant contribution to risk. The dominant accident sequences found by SNL are summarized and their characteristics presented in Table 6-1.

Calculations of the containment response to the accident sequences given in Table 6-1 were made by Carlson et al. (1982). The calculations were based on the MARCH/INTER code package and the results were used to quantify risk. The study by Levy et al. (1981) used the SNL results but declined to give the various sequences numerical values. This choice was probably made with good reason. Reilly et al. (1982) repeated calculations of containment response for a limited set of the SNL sequences using a MARCH/CORCON code package. With the differences between INTER and CORCON, one would expect calculable differences in risk between the SNL and INEL calculations.

The sequences listed in Table 6-1 show that the dominant ones result in or are a result of the following:

- Failure to remove decay heat from the reactor core.
- Loss of primary water and failure to provide make-up.
- Loss of long-term heat sink.

For ice condenser containments, the dominant failure mode resulting from one of the above three losses is failure of the containment from a pressure spike produced by hydrogen burning. This pressure spike may occur following the release of substantial amounts of fission products into the containment volume (S_1 and S_2 type sequences) and before vessel melt-through. It is important to note that these calculations assumed that the containment was not inerted and that igniters were absent. This aspect should be studied further because of the dominance of the γ failure.

Levy et al. (1981) argue that overpressurization from the rapid generation of steam resulting from the molten core falling into water in the reactor cavity will not be as important as in the large dry containments because the ice will condense the steam. This will not be the case for S_1 and S_2 sequences because the steam from the primary system will probably melt the ice. The result may just be a shift from the γ failure to the S_1 or S_2 failure. Furthermore,

TABLE 6-1. DOMINANT ACCIDENT SEQUENCES AND PROBABILITIES FROM THE SANDIA SEQUOYAH STUDY

Sequence	Description	Type of Failure	Timing of Failure	Type of Radio-Active Release	Probability	Fractional Contributions
S2HF-γ	Failure of the emergency core cooling recirculation and CSRS given a small LOCA (1/2" < D < 2")	γ - pressure spike from hydrogen burning	Prior to vessel melt through	RSS category PWR-2	5.4×10^{-6}	0.093
S1HF-δ,γ	Failure of the ECCRS and CSRS given a small LOCA (2" < D < 6")	γ - pressure spike from H_2 burning	Prior to vessel melt through	RSS category PWR-3	2.9×10^{-6}	0.050
		δ - slow overpressurization due to noncondensibles	After vessel melt through			
S2H-γ	Failure of the ECRS given a small LOCA (1/2" < D < 2")	γ - pressure spike from H_2 burn	Prior to vessel melt through	RSS category PWR-3	1.7×10^{-5}	0.293
S1H-α	Failure of the ECRS given a small LOCA (2" < D < 6")	α - in vessel steam explosion	Prior to vessel melt through	RSS category PWR-1	1×10^{-7}	0.002
TML-γ	Failure of the Power Conversion System and Auxiliary Feedwater System given a transient event	γ - hydrogen burn following core/concrete interaction	After vessel melt through	RSS category PWR-3	3×10^{-6}	0.054
S1H-γ	Failure of ECRS given a small LOCA (2" < D < 6")	γ - hydrogen burn induced spike	Prior to vessel melt through	RSS category PWR-4	1×10^{-5}	0.180
S2D-γ	Failure of the ECIS given a small LOCA (1/2" < D < 2")	γ - hydrogen burn and overpressurization	After vessel melt through	RSS category PWR-4	6×10^{-6}	0.108
S1D-γ	Failure of ECIS given a small LOCA (2" < D < 6")	γ - hydrogen burn and overpressurization	After vessel melt through	RSS category PWR-5	4×10^{-6}	0.072

TABLE 6-1. DOMINANT ACCIDENT SEQUENCES AND PROBABILITIES FROM THE SANDIA SEQUOYAH STUDY (CONCLUDED)

Sequence	Description	Type of Failure	Timing of Failure	Type of Radio-Active Release	Probability	Fractional Contributions
V	Rupture of the LPIS safety valves	LOCA outside of containment			4.6×10^{-6}	0.079
TLMB-α,δ,γ	Failure of PCS and AFW given loss of all power and no recovery	α δ γ	After vessel failure	RSS PWR-1 RSS PWR-2 RSS PWR-2	3.8×10^{-11} 2.0×10^{-6} 7.0×10^{-7}	0.0 0.036 0.017
$S_2C-\delta$	Failure of CSIS given a small LOCA (1/2" < D < 2")	δ - slow over-pressurization	After vessel failure	RSS PWR-3	$< 10^{-7}$	0.005
				Total	5.6×10^{-5}	1.00

Source: NUREG/CR-1659.

DEFINITIONS:

Initiating Events

S_2 - SBLOCA (1/2" < D < 2")
S1 - SBLOCA (2" < D < 6")
V - Interfacing systems rupture
T - Transient loss of AC power

System Failures

H - ECCRS
F - Containment spray recirculation systems
M - Power conversion system
L - Auxiliary feedwater system
D - ECCIS

the water in the reactor cavity may be very cold, thereby amplifying uncertainties about ex-vessel steam explosions.

All three studies ignore loss of all site power (the TLMB sequence). The reason for its small contribution to risk is given by SNL as being a consequence of the high reliability of the Tennessee Valley Authority (TVA) grid. Some areas may not have such reliable off-site power and as such may need a reevaluation.

During the NRR review of floating nuclear power plants (FNPs), the V sequence probability was reduced on the basis of arguments that leak testing could be effective. In some cases, leak testing before failure may be effective. Here it appears that periodic leak testing of the type described could be more harmful than helpful. Occasional water hammers resulting from the leak-test procedure may damage the valves while determining whether they leak at the end of a given test interval. Increasing the V sequence probability to its earlier value may significantly shift the value/impact evaluations that were calculated without specifying a fix.

The β mode, containment leak tightness, is essentially ignored in the three studies. The history of leak testing should be enough to call our attention to this mode of failure as one deserving early attention, along with the V sequence, in any mitigation study.

6.3 BENEFITS FROM MITIGATION SYSTEMS INSTALLATION

Studies of severe accident mitigation in ice condenser containments have been few, with the exception of the hydrogen-controlled burning studies for Sequoyah-1. The study by Levy et al. (1981) sets down a basis for mitigation or prevention but actually infers similarity to BWRs in order to reach conclusions. The more recent study by Reilly et al. (1982) is much more complete but still is based on the RSSMAP studies. Many aspects of mitigation and prevention in ice condenser containments are similar to large dry containments. The reader is referred to Section 7.3 for additional information. In this section, the results of Reilly's work will be used to illustrate the potential range of benefits achievable in ice condenser containments.

6.3.1 Potential Risk Averted

There are three obvious approaches, all of which are needed, to reduce risk in a plant with an ice condenser containment:

- Control hydrogen burning to avoid a pressure spike.

- Reduce noncondensible gas generation (H_2 and/or CO_2 from concrete decomposition) or vent to avoid overpressurization.

- Remove thermal energy to a dedicated long-term heat sink or vent to avoid overpressurization.

Note that reduction of the noncondensible gas generation also reduces the potential for basemat penetration (ϵ failure mode). Not only are these objectives the same as in all other LWRs (with each one yielding different risk reduction returns), but as pointed out by Boyd et al. (1981) ice condenser plants seem to have approximately the same risk as other commercial LWRs. As noted by Reilly et al. (1982), without a well-defined safety goal (and the means to assess it), it is not clear that the relatively higher Class 9 accident risk is unacceptable. The factor of several hundred in increased risk resulting from Class 9 accidents relative to other risks provides a challenging goal for either prevention or mitigation.

The risk that could be averted is determined from the dominant sequence frequencies, the conditional probability of a given containment failure mode, and the consequences. The result for Sequoyah-1 is shown as a function of release category in Table 6-2 and containment failure mode in Table 6-3, taken directly from Reilly et al. (1982). Table 6-4 shows these same results as contributions by the dominant sequences.

Examination of Table 6-4 clearly shows that control of hydrogen to avoid a pressure spike has the greatest potential for risk reduction. This rather unsurprising result has already led to installation of igniters being put in place at Sequoyah-1 and a plan for their use in GESSAR. It is not clear at this time why there is only property damage reflected in Tables 6-3 and 6-4 for cases where γ does not occur. Further review will be necessary to clarify this point.

6.3.2 Value of Mitigation

The potential for mitigation in terms of public risk can be seen by estimating the risk following elimination of the following six failure modes and sequences:

TABLE 6-2. EXPECTED CONSEQUENCES PER RELEASE - NORTHWEST RIVER VALLEY COMPOSITE SITE*

Release Category	Early Fatalities	Latent Cancer Fatalities/Yr	Property Damage* (10^6 $)
1_a***	91	120	2050
1_b***	8	114	2270
2	7	67	2440
3	0.4	55	987
4	0	18	335
5	0	6	201
6	0	1	173
7	0	0	171
8	0	0	1
	106.4	381	8628

Source: Reilly, 1982.

*Based on RSS source term.

**1974 dollars.

***Accident releases in Category 1 have two distinct energy releases that affect consequences. Category 1a involves an energy release of 20,000,000 Btu/hr and (4/9) of the total probability of Category 1. Category 1b involves an energy release of 520,000,000 Btu/hr with (5/9) of the total probability. These were combined to obtain an average value for Category 1 releases for this study.

Basis: Work 1400 Source Terms and Methodology.

TABLE 6-3. CONTRIBUTIONS OF CONTAINMENT FAILURE MODES TO EXPECTED RISK BEFORE MODIFICATION*

Containment Failure Mode	Early Fatalities/Year		Latent Fatalities/Year		Property Damage ($1000/yr)	
	Expected Risks	Percent of Total	Expected Risks	Percent of Total	Expected Risks	Percent of Total
α	2.38(7)**	0.3	6.18(7)	~0	0.01	~0
(early failure)	8.40(7)	1.0	1.16(4)	4.8	2.1	3.7
γ,δ	1.22(6)	1.5	1.65(4)	6.8	3.0	5.3
γ	4.58(5)	57.0	1.84(3)	75.7	40.3	71.2
ν	3.22(5)	40.1	3.08(4)	12.7	11.2	19.8
	8.03(5)		2.43(3)		56.6	
(if does not occur)	0	—	0	—	7.3	30.9

*Based on RSS source term.
**That is, 2.38×10^{-7}.

Source: Reilly, 1982.

TABLE 6-4. CONTRIBUTIONS TO TOTAL RISKS BY DOMINANT PROBABILITY SEQUENCE IN EXPECTED RELEASE CATEGORIES BEFORE MODIFICATION

Sequence[a]	Risk Contributions[a]			Percentage of Total Risks		
	Early Fatalities/Year	Latent Cancer Fatalities/Year	Property Damage ($1000/yr)	Early Fatalities/Year	Latent Cancer Fatalities/Year	Property Damage ($1000/yr)
$S_1H-\alpha$	5.84(8)[b]	1.52(7)	~0.0	0.1	~0.0	~0.0
$S_2HF-\gamma$	3.78(5)	3.62(4)	13.2	47.1	14.9	23.3
$S_1HF-\gamma\delta$	1.16(6)	1.60(4)	2.9	1.4	6.6	5.1
$S_2H-\gamma$	6.80(6)	9.35(4)	16.8	8.5	38.5	29.6
$T_{2,3}ML-\gamma$	1.20(6)	1.65(4)	3.0	1.5	6.8	5.2
$S_1H-\gamma$	0.00	2.34(4)	4.4	0.0	9.6	7.7
$S_2D-\gamma$	0.00	1.13(4)	2.1	0.0	4.7	3.7
$S_1D-\gamma$	0.00	2.04(5)	0.7	0.0	0.8	1.2
V	3.22(5)	3.08(4)	11.2	40.1	12.7	19.8
$S_2G-\delta$	4.40(7)	6.05(5)	1.1	0.5	2.5	1.9
$S_1G-\delta$	2.36(7)	3.24(5)	0.6	0.3	1.3	1.0
$S_2H-\delta$	0.00	0.00	2.91	0.0	0.0	30.6
$TML-\delta$	0.00	0.00	0.51	0.0	0.0	5.4
$S_1H-\delta$	0.00	0.00	2.22	0.0	0.0	23.3
$S_2D-\delta$	0.00	0.00	1.08	0.0	0.0	11.3
$S_1D-\delta$	0.00	0.00	0.58	0.0	0.0	6.1
S_2HF and S_1HF unchanged.						

Source: Reilly, 1982.

[a]Based on RSS source terms and methodology.

[b]That is: 5.84×10^{-8}.

1. The interfacing valve sequence (V).

2. The common mode failure of ECC recirculation and spray recirculation (HF) due to plugging of upper compartment drains.

3. Uncontrolled hydrogen burning (γ).

4. Failure of containment heat removal (G).

5. Prevent or greatly reduce the probability of slow overpressurization (δ_2).

6. Prevent or greatly reduce the probability of basemat penetration (ϵ).

The probability of a steam explosion (α) was taken by Reilly to be 10^{-4}. This reduces its risk contribution to an almost negligible value. Table 6-5 summarizes the results in terms of risk reduction factors.

Examination of Table 6-5 shows that for the Sequoyah PWR ice-condenser containment the greatest potential for risk reduction lies in mitigating the hydrogen burn (or failure mode). Depending on the measure used, the risk reduction factor is about an order of magnitude (8.7 for early fatalities, 13.3 for latent fatalities). Because loss of off-site power was not considered important at this plant, the results in Table 6-5 may not apply to other ice condenser containments.

TABLE 6-5. POTENTIAL RISK IMPROVEMENT FACTORS FOR ICE-CONDENSER PLANT (RATIO OF RISK WITHOUT IMPROVEMENT TO RISK WITH IMPROVEMENT)*

Improvement (i.e., Sequence or Failure Mode Eliminated)	Early Fatalities	Latent Cancer Fatalities	Property Damage
HF(common-mode)	1.9	1.3	1.4
V	4.4	1.2	1.4
Y	8.7	13.3	3.1
G	2.6	4.4	1.2
Late release; i.e., δ (release Category 7)	1.0	1.0	16.3
Total potential improvement factor	190.0	90.0	120.0

*Based on RSS source term estimates.

A basic weakness in the approach becomes apparent when results for the large dry PWR containment are compared with those from the work shown here. The large dry PWR effort was based on the Zion plant study which has received the attention of several national laboratories as well as industry and the NRC. The ice condenser work was based on Sequoyah-1 results coming out of the RSSMAP work and does not include external events. The numbers are difficult to relate to one another in a meaningful way. Site-specific initiators (external events such as seismic and loss of site power) need to be included in any consideration. Hence, the results presented here should not be compared with those presented for large dry containments.

6.4 MITIGATION OPPORTUNITIES FOR ICE CONTAINMENT

The maximum consequences and subsequent risk arise from accident sequences leading to core melt and containment failure. How the accident sequence proceeds to core melt and containment failure will vary considerably from one type of accident to another with a few dominant accidents that can be specified. Once the dominant accident sequences are identified, one can begin to formulate strategies to reduce consequences in ice condenser containments.

In the previous section it was noted that the risk at Sequoyah is mostly due to the following sequences:

- S_2HF - (γ,δ)
- S_2H - (γ)
- V

A study of the tables in the previous section will reveal that the greatest part of the risk is due to the V-sequence and uncontrolled hydrogen burning (γ) when classified according to failure mode. The biggest contributor to the risk in the S_2HF sequence is the common mode failure resulting from plugged drains. It is not possible to fully prevent the V sequence. One could contain the core melt and possibly do something to the secondary containment to prevent the escape of radionuclides. Such considerations are beyond the scope of this work. Most suppression-type containments are inerted or have igniters to reduce the possibility of a sudden hydrogen burn. The igniters, even though now installed, are still being studied and the final word on their effectiveness is at least a year away. Unlike large dry containments, a core melt can lead to containment failure even though the containment cooling system is operable

because the volume is small enough for noncondensibles to have an important effect.

Accident sequences like TLMB or TLMB' where the initiator is a loss of power were found to be relatively unimportant to risk at Sequoyah because on-site power unavailability was low. When other sites are considered, this may not be the case. Furthermore, when the V sequence is assumed to be mitigated by some means and the igniters are assumed to be effective, a whole host of other sequences becomes important. The lower containment design pressure and smaller volume may give the δ_2 failure modes (slow overpressurization) more risk potential. In particular, circumstances where the ice is melted early in the accident sequence make the ice condenser particularly vulnerable.

6.4.1 Cumulative Requirements for Ice Containment Mitigation Systems

To accomplish reliable mitigation, the accident end-state must be accommodated in a manner that leaves only a predictable sequence of events and prevents these from leading to a failure. Conditions existing at the initiation of a core-melt accident are the same as those cited earlier for other suppression-type containments. They are briefly repeated here for completeness.

1. All electric power and most controls and instrumentation (except R.G. 1.97 instrumentation) has been lost.

2. The ice in the ice condenser has melted and affords no further heat sink capability.

3. All core cooling systems are inoperable.

4. All normal containment heat removal systems are inoperable.

5. All steel in the path of the melting core will accompany the core debris into the lower reactor cavity.

6. Both high-pressure/high-temperature and low-pressure/low-temperature core debris will be considered.

7. Much of the Zircaloy cladding will be assumed to have reacted with steam to form hydrogen gas.

8. It is assumed that plant operators will not contribute to mitigation operation.

The functional requirements for a mitigation system are noted below:

1. The mitigation system must perform in the environment described above.

2. Proper combustion of hydrogen must be assured or the environment must be inerted and overpressure controlled as a result of additional mass to the containment volume.

3. Sufficient long-term heat sink availability must be assured.

4. Noncondensible gas generation from core/concrete interactions must be minimized or its effect on containment pressure must be controlled.

5. Flooding of the reactor cavity must be controlled to preclude overpressurization resulting from rapid generation of steam immediately following vessel failure.

The criteria for choosing a mitigation scheme were presented in detail in Section 2 and are not repeated here.

6.4.2 Candidate Means for Meeting Requirements

Mitigation devices and components useful for ice condenser containments will serve the same function as those for other suppression-type containments (Mark I, II and III BWRs). Mitigation systems appropriate for use in ice condenser containments are reviewed briefly in the companion Task 2 report. The following discussion should be considered as a supplement to that found in Section 3.4 of this report. To structure this section, the following functional requirements are considered:

1. Hydrogen control.

2. Control of flooding.

3. Basemat protection.

4. Overpressure protection.

5. Long-term heat removal.

Hydrogen Control

There are a number of possible approaches for limiting the consequences of hydrogen generation and burning. They include preinerting of the containment (as done for the BWR Mark I), inerting of the containment after the accident but before formation of significant amounts of hydrogen (post-inerting), and the controlled local burning of hydrogen using distributed ignition systems (Sequoyah).

Preinerting of the containment volume for hydrogen control is done in most Mark I containments and is a mature system that is easily employed. A major disadvantage of its use is decreased access to equipment inside the containment. The large volume of ice condenser containments and the additional equipment within the primary containment accentuate this shortcoming. It is the opinion of the industry that this shortcoming is not balanced by increased safety resulting from preinerting. An aspect of safety not taken into account by the industry is the reduction in the β failure mode where something is left open. An inerted containment will surely be much more leak-tight than otherwise might be expected. This aspect needs some attention before preinerting is discarded.

Postinerting (or the prompt introduction of an inert gas into the containment volume after an accident is underway to preclude hydrogen burning) is advantageous to ice condenser containments because it preserves access to equipment in normal operation. There are, however, a number of potential problems needing solutions before postinerting can be considered viable. The most important is the addition of even more gas mass to the containment volume when the pressure may already be critical. By the time enough gas is added to effectively inert the containment volume, overpressure from other sources would be greatly aggravated. A CO_2 postinerting system is being developed by GE for the Allens Creek plant (a Mark III BWR).

Controlled burning of hydrogen using a distributed ignition system (DIS) is the approach receiving the most favor at the EPRI workshop on hydrogen control (Boyd, 1981). The large containment volumes (1,250,000 ft^3) and amounts of hydrogen involved provide a great deal of incentive for burning in place. The DIS is being developed by TVA for Sequoyah. Igniters are in place, and testing programs are underway to ensure that they perform as planned under a variety of conditions including sprays. Recent studies by Cybulkis (1983) led him to conclude that igniters may not be so effective as once believed. Assuming that the igniters had an 8 percent burn point, he shows that the pressure produced

could still lead to containment failure. He did conclude, however, that we are probably better off with them than without them. Sufficient work is underway to yield proof (or disproof) of the concept.

Other postinerting systems based on Halon gas, foam generation, or oxygen consumption have also been considered.

Control of Flooding

The α failure mode (steam explosion) does not contribute significantly to the calculated risk when $P(\alpha) < 10^{-4}$. This value was based on extrapolation of small-scale tests with simulant materials. Recent experiments at SNL cast some suspicion on the earlier conclusions and although $P(\alpha)$ is still thought to be low, it is not zero. Prudent practice would therefore dictate that the large-scale interaction of molten core materials and water be avoided because of the possibility of both steam explosions (α) and steam spikes (δ_1). It is not clear what one can do to avoid steam explosions. A good mitigation scheme should, however, limit the amount of water in the cavity at the time of vessel failure.

Basemat Protection

Molten core materials will thermally attack the concrete basemat, liberating large amounts of noncondensible gases while generating large amounts of aerosols. Basemat protection from this attack could be achieved in several ways. Either the flooded pebble bed, which is essentially a passive device, or the dry crucible core catcher, could provide this protection. Properly designed, the basemat protection system would prevent or greatly reduce the probability of a large-scale molten core debris/water interaction--a steam spike--when the vessel fails as well preventing basemat attack. This would eliminate, or at least significantly reduce, Category 7 releases from the γ and δ_2 failure modes and possible high-consequence early failures by γ or δ_1 modes.

Overpressure Protection

If flooding of the reactor cavity is controlled and basemat attack is prevented, late overpressurization should not occur. If, however, one or both of those functions were not provided or were to fail, one should be able to deal separately with overpressurization. A small vent, capable of about 10,000 cfm, would probably be sufficient. In some cases, the SGTS is adequate.

Long-Term Heat Removal

Adequate long-term containment heat removal is essential to any mitigation system whose purpose is to completely eliminate releases of radioactivity. A number of alternatives have been considered, and three are discussed in Section 3.4 of this report. An ice condenser containment, however, presents some differences.

If one were, for example, to use dedicated sprays, their location is critical. Locating them in the upper compartment subjects them to the common mode failure that is a major contributor to core melt. On the other hand, locating them in the lower compartment may not be as effective at condensing steam and the environment would be harsh. Vacuum breakers between the upper and lower compartments may be a solution to part of the problem.

The deep crucible core catcher has its own heat removal system. Problems with designing a "sure" system for the two-compartment containment may lead to favoring the deep crucible over the heat removal systems. Of course, an alternative is always two systems--one for the upper compartment and another for the lower. Cost will play a role in these decisions.

Appropriately selected components from the above categories along with certain other design modifications to ensure adequate protection of containment penetrations and other such considerations would adequately mitigate the consequences of a severe accident in a PWR ice condenser containment. Before a set of devices is selected, certain risk studies are needed. The question of cost versus risk averted with the V-sequence and a failure mode eliminated needs addressing with particular attention to TLM-type initiators using data from plants other than Sequoyah. The performance of the chosen "set" of components must be analyzed for cost for risk-averted information.

The SASA program is devoting a portion of its efforts to PWR systems. Within this exercise, both the large dry and ice condenser plants will receive attention. As with SASA efforts that have largely been completed for Mark I BWR plants, the objective will be to establish the adequacy of plant equipment, instrumentation, and operator training. This review will focus on those dominant accident sequences leading to core-melt incidents. Recommendations will be made to improve the effectiveness of plant systems and operator actions in order to prevent core melt.

The results of the SASA studies related to PWRs are not yet available. When they are, it will be possible to compare their cost and effectiveness in preventing core melt with the mitigation system characteristics described above. It will be necessary, however, to take the SASA results and to compare the risk reduction parameters.

6.5 SUMMARY

This review of severe accident responses in an ice condenser containment was made particularly straightforward by the existence of the Sequoyah RSSMAP study (Carlson et al. 1981), the Reilly et al. (1982) study of a core-melt mitigation system for Sequoyah, and the Levy et al. (1981) study of suppression containments. A very good summary is given by Reilly et al. (1982) and reproduced here with some modification.

The ice condenser containment risk studies yield sequences similar to these for the large dry containments, but show that the consequences are very sensitive to early hydrogen burning. Another difference is the common mode failure resulting when the drains from the upper compartment to the lower compartment become plugged. The two-compartment containment leads to a need for a different long-term cooling system design than the simple drywell spray-suppression pool cooling system suggested for the BWR containments.

Mitigation of severe accidents in ice condenser containments can be summarized by stating the functional requirements:

1. Hydrogen control.
2. Reactor cavity water control.
3. Basemat protection (or overpressurization protection).
4. Long-term cooling.

These functional requirements are listed in their apparent order of importance to risk reduction even though not separable.

The work by Reilly et al. clearly shows that significant risk reduction can be achieved for relatively little cost. Without the gas combustor, their estimated cost for water control in the reactor cavity, pebble bed basemat protection, and late venting is under $\$2 \times 10^6$. Mitigation devices are available to meet the requirements, but questions both of cost and of compatibility with existing systems need

further exploration. When site-specific studies are available, the value associated with risk reduction can be determined and compared with cost.

CHAPTER 7. PWR DRY CONTAINMENT

The term dry containment refers to the absence of a water pool within the containment structure. The Mark I, II, and III plants all include large suppression pools, and the ice condenser plants have large bins of ice. The dry containment houses the greatest number of U.S. nuclear reactor plants and is more diverse in form than the other major containment types. As shown in Appendix D, a total of 83 plants fall into this containment category (including, as a subset, the subatmospheric types). The diversity in size, configuration, and choice of construction materials is accounted for by the large number of different constructors and by the broad range of reactor power level of these plants. The electrical power outputs of the plants range from a low of 50 MWe at La Crosse to a high of 1285 MWe at the two Bellefonte plants in Corinth, Mississippi.

The seven subatmospheric containment plants listed in Appendix D are included as a subset of the dry containment class. The operational features of these containments and their complementary systems are sufficiently similar to the other dry containments to allow their inclusion. The normal containment operating pressure of 9 to 11 psia is slightly less than the atmospheric pressure of the other dry containments. This fact has several ramifications with respect to structural design and containment leakage as will be discussed below.

Early dry containments employed steel spheres (e.g., Yankee Rowe and Big Rock Point). As the ratings and sizes of the dry containments grew, the use of steel containments was suspended for some time (Mehta, 1977). Concrete containments became more common. The concrete shell was relatively inexpensive and provided missile and radiation shielding during normal and accident conditions. The ability to withstand internal pressures was achieved through the use of reinforcing steel. Subsequently, the steel design was revived and applied in both the large cylindrical and large spherical forms. However, with these bigger plants the steel primary containments are completely surrounded by reinforced concrete secondary containments. The secondary structures provide missile protection and radioactivity shielding. Secondary containments for nonsteel primary containments are uncommon. Only six of the 68 dry containments not having a steel primary containment have a secondary containment (which is steel in three cases and reinforced concrete in the other three cases).

The Mark I, II and III plants contain exclusively GE BWRs (46 plants), and ice condenser plants contain exclusively Westinghouse PWRs (10 plants). Dry containments, on the other hand, contain a variety of reactor types made by a variety of manufacturers. Nuclear steam system suppliers for dry containments include Combustion Engineering (18 plants), Babcock & Wilcox (13 plants), Westinghouse (48 plants), and GE (2 plants).

The characteristics of dry containment--including subatmospheric types--are discussed in further detail in Section 7.1. The containment is an Engineered Safety Feature of the plant and, because of this, is designed to more stringent standards than other plant features. The DBA set provides an envelope of severe accident environments that the Engineered Safety Features must accommodate without adverse radiation release from the plant (see Appendix B). Other Engineered Safety Features directly complement and make possible the successful operation of the containment. Complementary Engineered Safety Features include the containment long-term cooling system, isolation features, the secondary containment, and the combustible gas control system. These systems are also discussed in Section 7.1, and reference is made to the DBA conditions they can accommodate plus other conditions they cannot.

PRA studies have highlighted accident sequences beyond the design basis that can cause containment failure. The dominant accident sequences leading to failure are identified in Section 7.2. Mitigation systems are then proposed in Sections 7.3 and 7.4 that may be able to alleviate--at least to some extent--the consequences of severe accidents (i.e., Class 9 accidents). The degree to which this is theoretically possible is estimated below along with the value of any risk reduction. The calculation of quantitative change in risk is made through modifications of the PRA results. PRA study results are available for the Surry, Oconee, Zion, and the Indian Point plants. It is assumed in Section 7.3 that certain release categories can be wholly eliminated through the action of selected ideal mitigation systems (later tasks will estimate the actual extent to which real-world mitigation systems can match this ideal performance). The value of risk reduction is computed on the basis of the change in man-rem caused by eliminating release categories. Based on the value of the various mitigation systems, recommendations are made for further detailed study of mitigation systems in future tasks (Section 7.5).

7.1 DESCRIPTION OF THE DRY CONTAINMENT AND RELATED SYSTEMS

The emphasis in this section will be on the nuclear reactor itself, the reactor internals, the ECCSs, the primary and secondary containment, and the Engineered Safety Features associated with the primary containment function. Items in the last category include the containment isolation system, the long-term containment heat removal system, and the combustible gas control equipment.

The reactor description highlights the materials potentially involved in a core meltdown. Of interest are the quantity of fuel, cladding, and steel available as meltdown constituents (the mixture is referred to as corium). The lower head, lower head contents, and lower head penetrations are relevant to the release of core materials from the vessel. The ECCSs are briefly described. However, in the balance of the mitigation system analysis it is assumed that the ECCS has failed to function for an unspecified reason or reasons.

A severe accident mitigation system is likely to depend--at least in part--on a fully functioning primary containment. Therefore, the containment dimensions, design pressure, construction materials, internal heat sinks and leakage rate are highlighted. The complementary systems needed to maintain containment integrity are also of interest. Severe accident mitigation will probably require containment isolation, heat removal, and combustible gas control. Descriptions of the existing plant Engineered Safety Features that are designed to accomplish these same functions under DBA conditions are included in this section. Reference is made to Appendices A and B for a review of the plant safety system strategy and DBA set.

7.1.1 Reactor and Primary System

U.S. nuclear power plants utilizing dry containments (including the subatmospheric types) have PWRs in nearly every case. Two exceptions to this general rule are the small BWR plants known as La Crosse and Big Rock Point (see Appendix D for a complete listing of all 83 plants assigned to the dry containment category). The nuclear steam supply system consists of the following major elements: the reactor pressure vessel, the coolant circulating pumps, the steam generators, the pressurizer, and the interconnecting primary system piping. Subcooled water circulates at a pressure level of about 2200 psig between the reactor core and the multiple steam generators. All of the steam supply components associated with the primary system are located within the primary containment structure.

In contrast with BWR pressure vessels, the reactor vessel in most PWR plants is supported at the upper head flange or at the main nozzles on the side of the vessel. BWR vessels are are supported by a circumferential skirt attached to the vessel head. The BWR vessel skirt is susceptible to thermally-induced damage because of its position in the event of a core meltdown accident. PWR vessels are likely to remain in place despite serious melting in the lower portions of the vessel. This feature helps prevent gross vessel motion from tearing out or rupturing piping penetrations in the primary containment.

Another basic difference in the PWR vessel from the BWR vessel is that all of the control rods enter from the top rather than the bottom. The lower vessel head in some PWRs is completely free of penetrations while in the majority of plants a large number of instrument guide tubes (diameter 1 to 2 inches) enter the vessel through the lower head. Figure 7-1 shows how these guide tubes are arranged in a typical plant. The instrument tubes lead from the lower head of the vessel down int the reactor cavity and curve into a passage in the containment basemat (referred to as the keyway) which takes them under the shield wall and upward to the seal table on the operating deck of the containment building. The large turning radius of the tubes results in a large open area below the RPV, but working in or using this space is complicated by the presence of these instrument tubes. The containment sump may be located in the keyway.

The size of the reactor vessel in a large PWR (1000 to 1300 MWe) is roughly 175 inches in diameter and 400 inches in length. The core diameter is approximately 130 inches in a large plant, and the number of fuel rods is approximately 50,000. The thickness of the steel reactor vessel is in the 5- to 9-inch range.

The quantity of uranium dioxide fuel in the core is typically 95 to 140 lb/MWt capacity of the plant. Furthermore, the mass of Zircaloy cladding is 0.33 to 0.43 times the mass of uranium dioxide depending on the plant of interest. A plant with a thermal output of 3800 MWt could have as much as 420,000 lb of fuel and 120,000 lb of cladding material. These quantities represent upper limits to the quantity of core material that may be involved in a core disruptive accident.

Pressurized, subcooled water at a temperature of about 550°F is circulated by multiple, electrically driven pumps between the reactor and the multiple steam generators. Feedwater is converted to steam in the generators for delivery to the

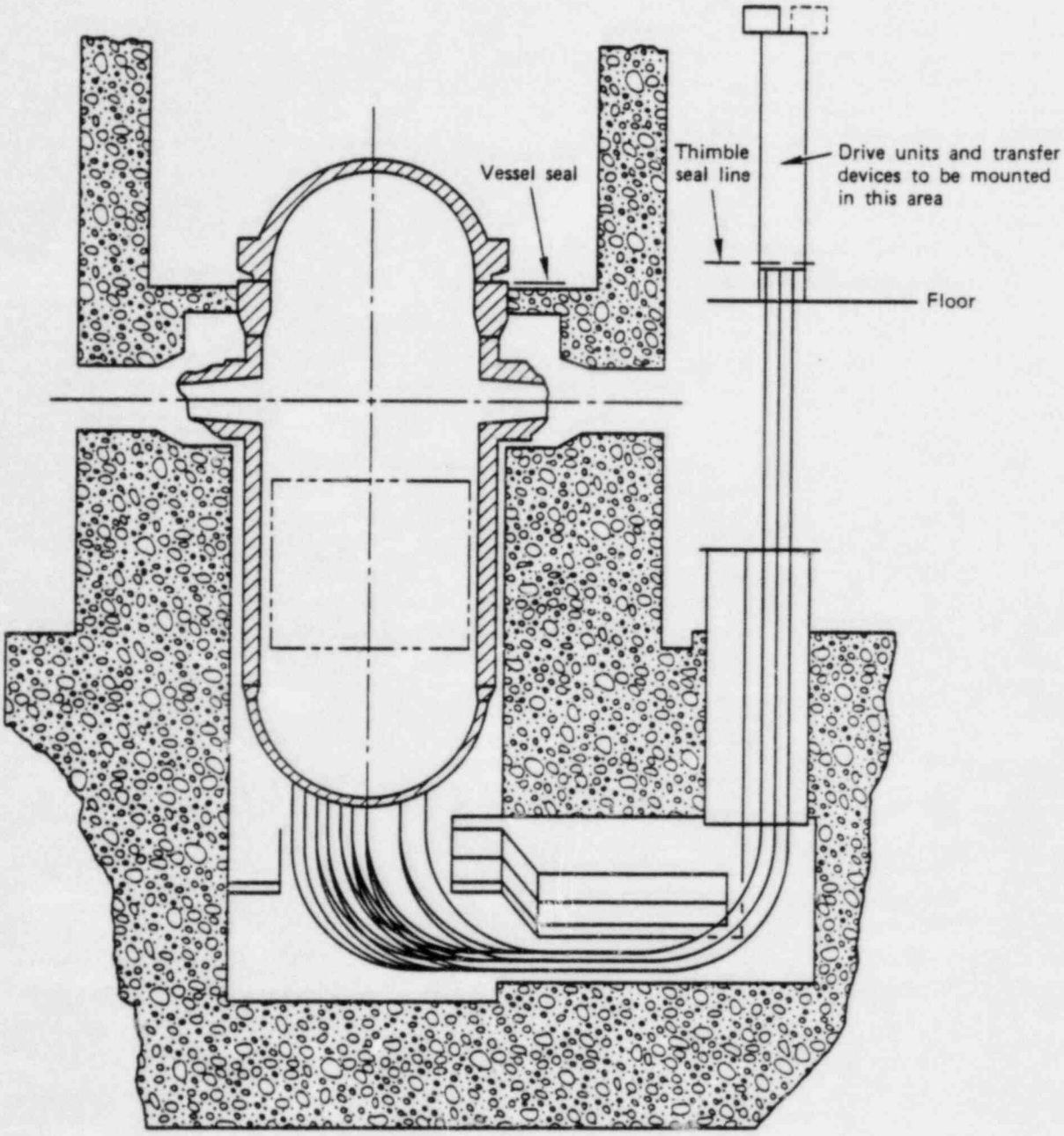


Figure 7-1. Prairie Island nuclear plant reactor cavity and in-core instrumentation tunnel.

turbine/generator set. An emergency feedwater supply system is available in the event primary feedwater is lost for whatever reason. A water-filled, electrically heated and spray-water-cooled pressurizer tank is used to control the operating pressure in the primary system. Water circulation through the core is at a rate of about 140,000,000 lb/hr in a 1200-MWe plant. Figure 7-2 shows the internals of a typical PWR vessel.

The ECCSs (or safety injection systems, SISs) are provided to limit the consequences of LOCAs in the primary system. Their function is to remove heat from the reactor core and thereby to minimize metal/water reactions and to prevent core degradation. Each element of the ECCS is an Engineered Safety Feature. The cooling systems are sized to accommodate primary system coolant losses due to (1) pipe breaks as large as the main circulation line, (2) control rod ejection, and (3) steam generator tube rupture. The ECCS is also able to accomplish reactor shutdown through the introduction of borated water to the reactor core.

Table 7-1 lists examples of three or four independent and functionally diverse ECCSs provided by the three U.S. PWR vendors. In a generic sense, the different vendors' systems are the same. The first element consists of multiple gaseous-nitrogen pressurized tanks of borated water. Borated water will be forced into the reactor vessel from these tanks if the vessel pressure drops below about 600 psig. Second are the high-pressure water injection pumps. In many plants these electrically driven pumps draw borated water from the refueling pool or from the borated water storage tank and can inject it at full reactor operating pressure into the primary system. Water can also be drawn by the high-pressure system from the containment sump. In some PWRs, the high-head pumps can take suction from the discharge of the low-head pumps when operating in the recirculation mode.

The third ECCS element--the low-pressure coolant injection system--operates in a fashion similar to the high-pressure system but at a lower pressure. It too can draw on the containment sump if necessary. The low-pressure system circulating water can be passed through heat exchangers in order to transfer shutdown and decay heat to the plant's ultimate heat sink. In some plants (e.g., Surry, Calvert Cliffs) containment heat is removed through heat exchangers in the containment spray system. Failure of the decay heat removal system constitutes one of the main contributors to the risk resulting from DBAs in dry containment reactors. Completely redundant, full capacity, high- and low-pressure

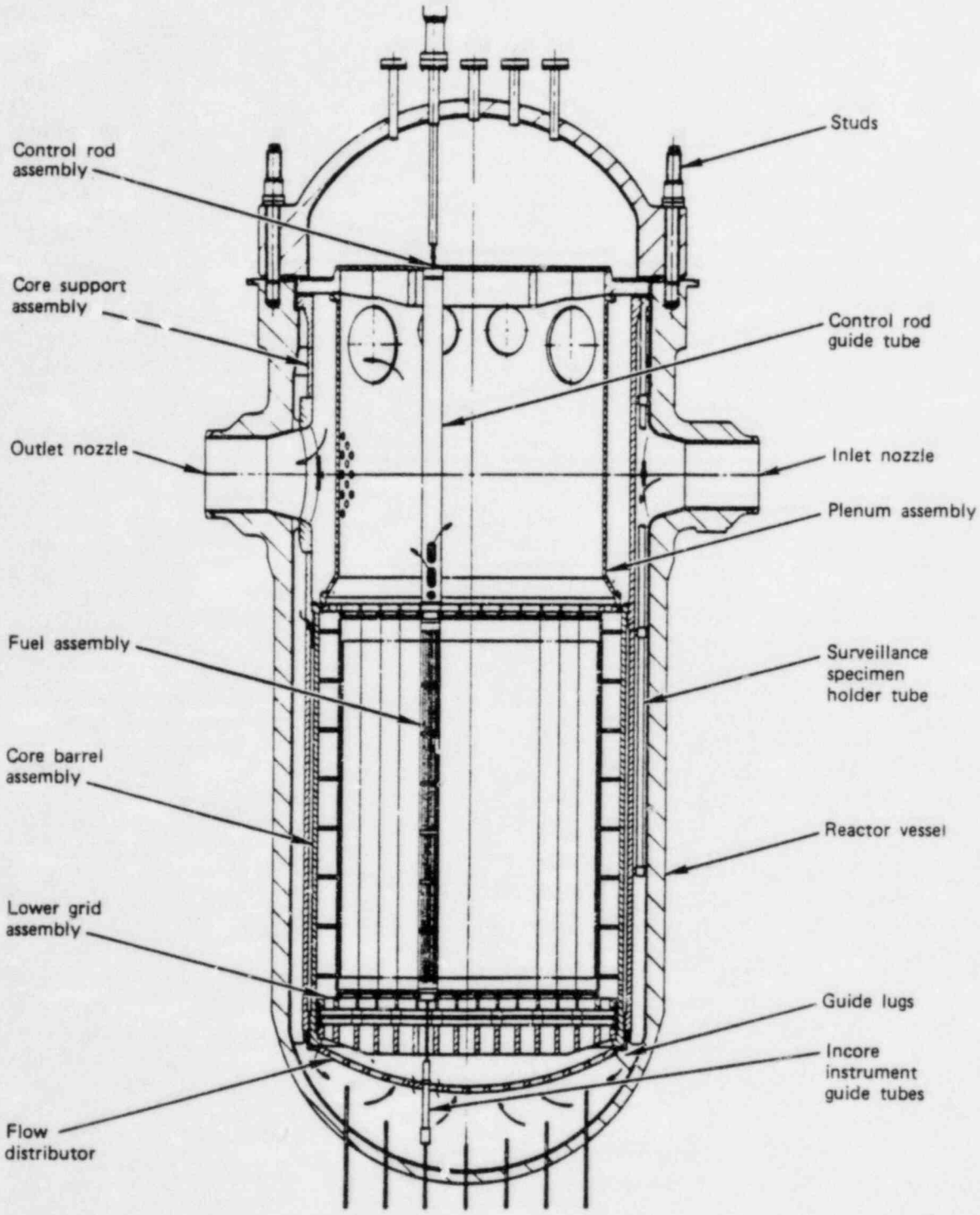


Figure 7-2. Reactor vessel and internals, general arrangement.

TABLE 7-1. PWR EMERGENCY CORE COOLING SYSTEMS

PWR vendor	ECCS
Babcock & Wilcox	Core flood (CF) High-pressure injection (HPI) Low-pressure injection (LPI)
Combustion Engineering	Safety injection tanks (SIT) Low-pressure safety injection (LPSI) High-pressure safety injection (HPSI)
Westinghouse	Accumulators Centrifugal charging pumps Safety injection pumps RHR low-pressure injection

injection systems are provided to maximize the likelihood of their successful operation.

7.1.2 Primary Containment

Appendix D lists 83 U.S. nuclear plants having dry containments. The total figure includes seven subatmospheric plants. All the containments enclose PWRs with the exception of the Big Rock Point and La Crosse units which are BWRs. The approximate volume (V) of the primary containments in all 83 plants is given by the expression $V = AP$ where V is expressed in millions of cubic feet. P is the plant's rated electrical output in megawatts. The constant A ranges from about 0.0021 to 0.0030 with a mean value of roughly 0.0025. A plant with an output of 1200 MWe has a containment volume of 3,000,000 ft^3 according to this expression. The design pressure for all the containments ranges from 11 to 60 psig (at a coincident temperature of about 275° F) with most plants in the 45- to 60-psig range.

Plants with dry containments rely primarily on a combination of strength and volume to retain the largest LOCA release considered in the DBA set. A second important function of the primary containment--in those plants lacking a secondary containment--is to shield the reactor and safety systems from external threats. Environmental threats include snow, wind, missiles, floods, and earthquakes. A third role of the containment is radiation shielding in the event of a LOCA. The importance of the ability to contain LOCA releases and shield equipment causes the primary containment to be classified as an Engineered Safety Feature. As such, it is subject to stringent design, materials, construction

and testing standards as called out in 10 CFR 50, NRC Regulatory Guides, as well as other codes and standards. The containment complements other safety features of the plant so that the maximum dose expected at the site boundary meets regulatory requirements under all design conditions.

While containments are referred to as being leak-tight, this is not actually the case. Design leakage rates are normally 0.1 percent of the containment volume per day at the design pressure and temperature. Periodic testing required by law is carried out to ensure that the containment leak-tightness is not lost. Operational experience has shown that this area needs attention and, as a result, it is now under serious scrutiny.

Three types of dry containment structures have been used to meet the stated requirements (Mehta, 1977). Briefly, they are:

1. Posttensioned, prestressed concrete with or without a secondary containment. Forty-six plants are in this subset (e.g., Palisades). Figure 7-3 shows some of the main features of this structural type.

2. Deformed-bar reinforced concrete with or without a secondary containment. Twenty-two plants are in this subset (e.g., Diablo Canyon). Figure 7-4 shows details of one such containment. All the subatmospheric plants have primary containments constructed in this manner.

3. Steel with or without a secondary containment. Fifteen plants are in this subset (e.g., Davis Besse). (See Figure 7-5 for pertinent details.)

The general characteristics of the reinforced concrete containments are noted as follows (Steigelmann, 1969):

1. Hemispherical dome.

2. Thickness of 3.5 to 4.5 ft for the wall and 2.5 to 3.5 ft for the dome.

3. Main reinforcing bars placed axially and circumferentially.

4. Seismic reinforcing bars placed helically at a 45° angle in both directions.

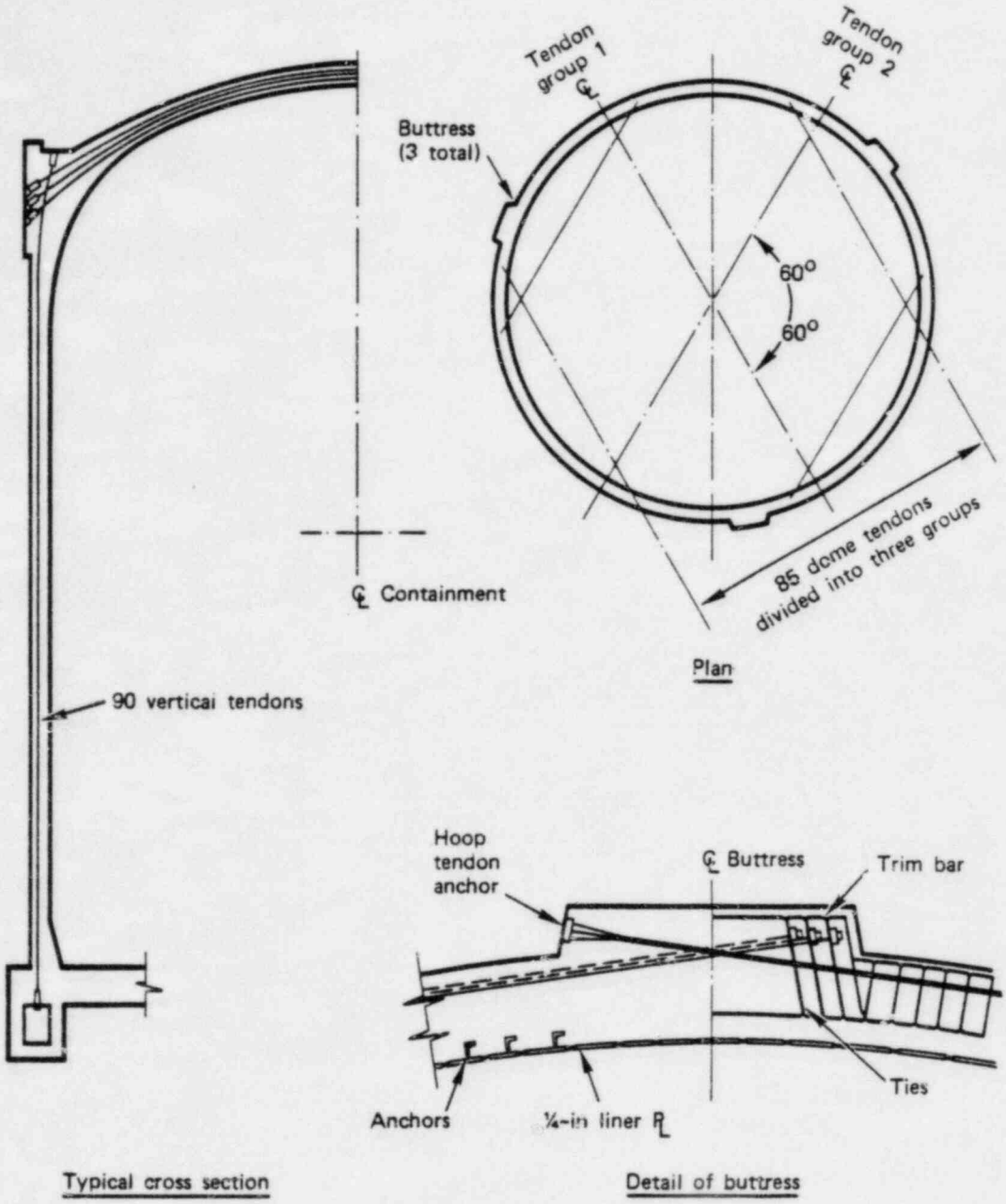


Figure 7-3. Prestressed concrete dry containment.
Source: Spencer 1982

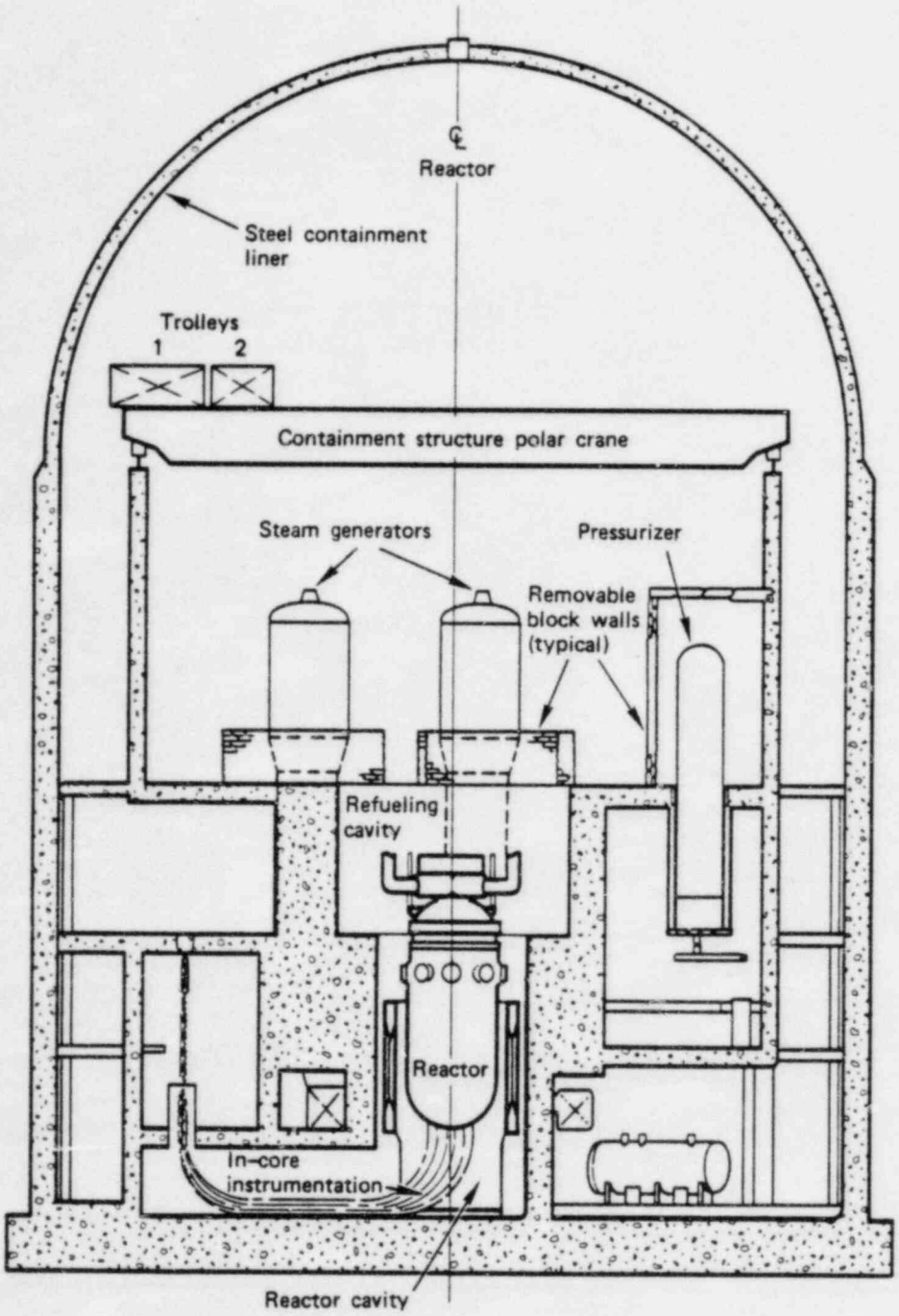


Figure 7-4. Reinforced concrete dry containment.
Source: Spencer 1982

5. Thermal insulation placed on the inside for the bottom 20 ft or so of the cylindrical wall portion. This minimizes thermal stress effects at the wall/basemat junction.

Generally the wall thickness of the reinforced-concrete containment is 0.5 to 1.0 ft thicker than that of a posttensioned concrete containment for a similar sized reactor. The internal design pressure is also generally lower, about 10 to 20 psig for similar sized reactors.

Each of the concrete containments is lined with a continuous 1/4- to 1/2-inch steel membrane for leakage control. The liner is anchored to the concrete by continuous angle, channel or discrete studs. All the plants are built on thick concrete basemats. Recent tests in the Federal Republic of Germany (FRG) have shown that the steel liners suffer rather large strains during a DBA. It is not clear that this has been properly factored into their design.

The subatmospheric containments are all built of reinforced concrete and do not have secondary containments. These primary containments normally are maintained at an operating pressure of 9 to 11 psia. Following a LOCA, the pressure is quickly returned to subatmospheric by the plant's ECCSs. This terminates containment leakage and provides a definite time limit for any potential off-site effects. In conventional containments, the post-LOCA containment pressure asymptotically approaches 0 psig. Table 7-2 illustrates containment pressures expected during LOCAs in two types of dry containments. Consequently, in the nonsubatmospheric plant a driving force for leakage may exist for an extended period. The need for a secondary containment in a subatmospheric plant is removed. The normal operating pressure in the subatmospheric plant is not so low that breathing apparatus would be required by workmen entering the containment.

Subatmospheric containments require adequate attention to ensure that in-leakage is controlled and that the negative pressure can be maintained. Otherwise the containment environment is not sufficiently isolated. With adequate maintenance, there is a reduced probability of a large leakage path following an accident. The structural requirements for the subatmospheric plants are not appreciably different from those of the more numerous reinforced-concrete atmospheric pressure plants. The latter types could be converted to subatmospheric operation with minor modifications (Noble, 1968).

TABLE 7-2. CONTAINMENT CONDITIONS IN ATMOSPHERIC AND SUBATMOSPHERIC CONTAINMENTS DURING A LOCA EVENT

	Subatmospheric Plant	Conventional (Atmospheric) Plant
Initial conditions (psig) @ 105°F		
Pressure	10.0	15.5
Water partial pressure	0.5	0.5
Air partial pressure	9.5	15.0
Peak containment pressure (50 psig)		
Temperature (°F)	285	273
Air partial pressure (psia)	12.4	19.6
Water partial pressure (psia)	52.3	45.1
Total pressure (psia)	64.7	64.7
Post-accident conditions		
Temperature (°F)	140	140
Air partial pressure (psia)	10.1	15.9
Water partial pressure (psia)	2.9	2.9
Total pressure (psia)	13.0	18.8
Total pressure (psig)	- 1.7	+ 4.1

Source: Noble, 1968.

The reduction in the amount of reinforcing required in a prestressed concrete containment allows the wall thickness to be less. Typically, the entire liner of the cylindrical wall is insulated with a PV foam plastic (covered by stainless steel sheet) as a way of limiting stresses and temperatures in the steel liner. The structural shape of the prestressed concrete containment is cylindrical with an ellipsoidal or hemispherical dome and concrete ring girder (a girder not required if the dome is hemispherical) at the wall/dome junction. Three to six buttresses run vertically on the outside of the containment wall. The hoop tendons are anchored at these buttress locations. Other prestressing tendons may run in vertical and diagonal directions.

A steel containment is normally supported on a flat concrete basemat. A typical cylindrical steel primary containment is 1.5 inches thick and has a 0.75-inch-thick hemispherical dome. Limiting the steel plate thickness to 1.5 inches eliminates the need for post-weld stress relieving. Steel containments are normally designed in accordance with the ASME Boiler and Pressure Vessel Code. Several

steel containments are spherical (La Crosse, Big Rock Point, San Onofre 1, Yankee Rowe, Yellow Creek 1 and 2) though most are cylindrical. Other than the two small BWR plants and the small Yankee Rowe plant, they all have concrete secondary containments for shielding protection.

The PWR vessel is closely surrounded with a concrete shield wall for biological shielding within the primary containment. Plant ventilation air is circulated upward through the annulus space between the vessel and the shield wall for cooling purposes. Figure 7-6 shows how the vessel can be positioned within the containment.

7.1.3 Secondary Containment

Relatively few of the dry containment nuclear power plants in the U.S. have secondary containments. Appendix D indicates only eighteen of the 83 plants classified as using dry containments (including subatmospheric types) have secondary containments. Eleven of these eighteen have steel primary containments while the balance have concrete primaries. The reinforced-concrete secondary containment is needed for biological and external missile shielding when a steel primary containment is used for leakage control. Plants with both concrete primary and secondary containments are Bellefonte 1 and 2, and Seabrook 1 and 2.

As with the ice condenser and BWR Mark I, II and III plants, the secondary containment provides an opportunity to capture and process leakage from the primary containment in the event of an accident that releases radioactivity to the containment. Following a plant accident, the 5- to 10-ft-wide annulus space between the two containments is maintained at a slightly negative pressure by fans discharging to the plant stack through filter trains. The annulus ventilation system (AVS) and filtration systems are Engineered Safety Features. Two full capacity systems are present, each electrically operated and actuated. The filtering action provided by the HEPA and the charcoal filters is intended to remove particulates and halogens from gases leaking through the primary containment. Actuation of the equipment is coincident with primary containment isolation.

The design pressure of the secondary containment is low--typically less than 1 psig. Leakage though the structure from the outside environment can be on the order of 50 to 100 percent of the annulus volume per day.

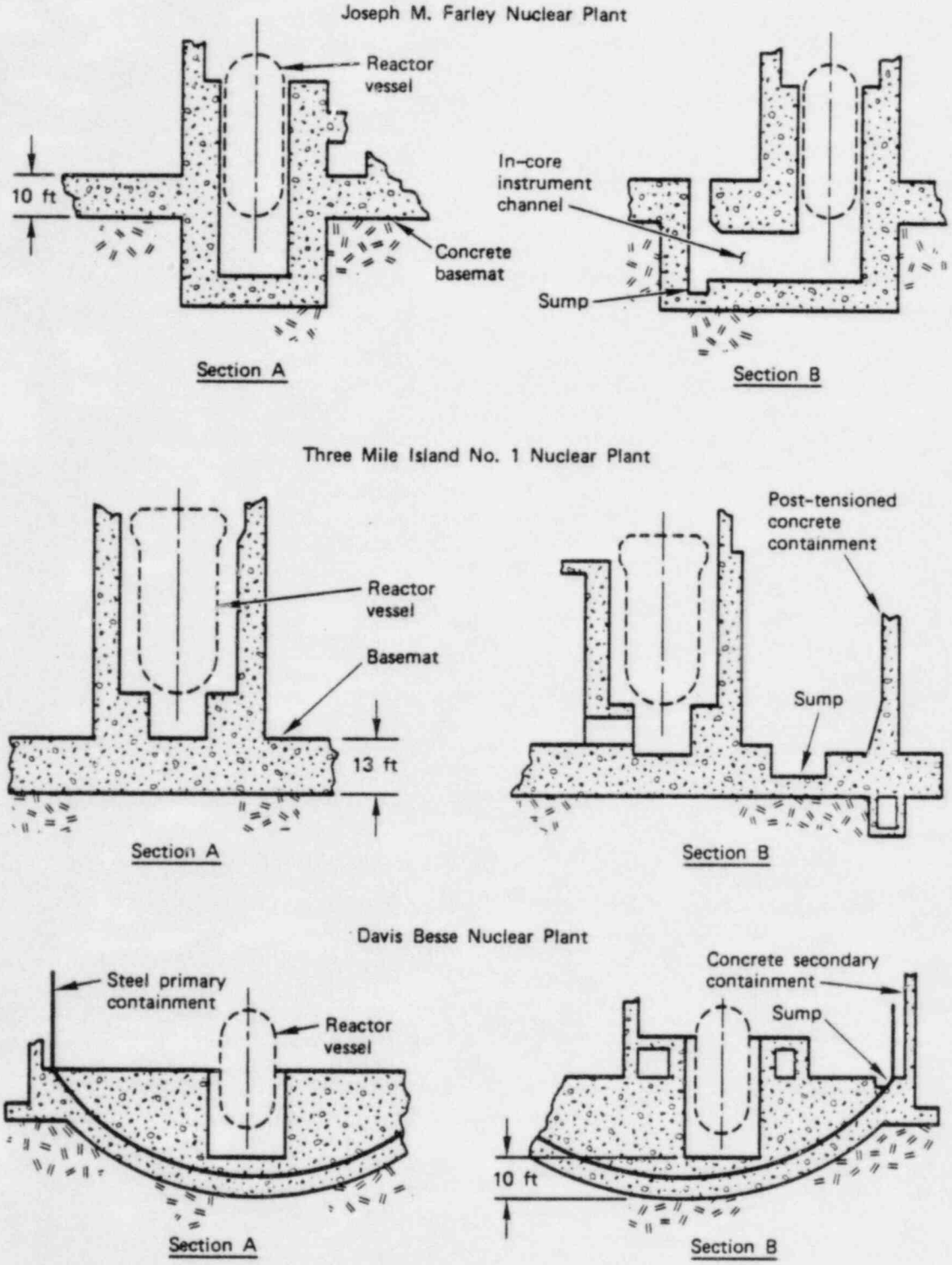


Figure 7-6. Variations in dry containment cavity configurations.

7.1.4 Containment Isolation and Protocol

The isolation of the primary containment penetrations is intended to help prevent the release of radioactivity in the event of a reactor accident such as a LOCA. Without this action, one of the main purposes of the containment would be circumvented. Isolation is effected automatically and/or manually through the use of a multiplicity of check valves, motor-operated valves, pneumatically operated valves, and hand block valves. This action is taken to maintain plant releases below statutory limits during DBAs and other, less severe, plant conditions.

The primary containment is penetrated at many locations. Penetrations are needed for equipment hatches, electrical power cables, instrumentation lines, personnel hatches, piping and ductwork. Hundreds of discrete items pass through the containment wall. Table 7-3 lists the piping penetrations in a typical PWR plant.

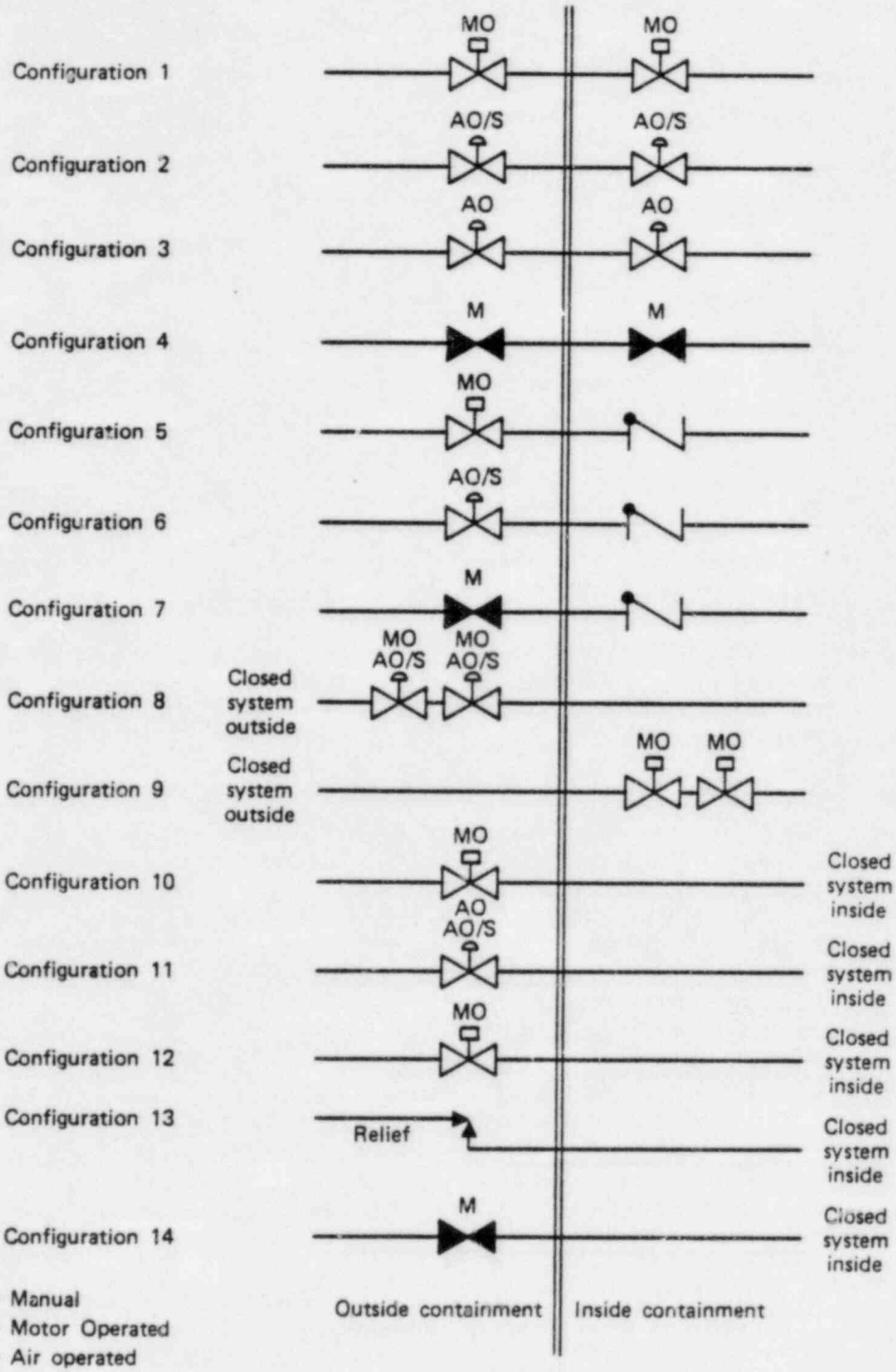
The containment isolation equipment is an Engineered Safety Feature and is designed to handle the extreme loads (temperature, pressure, humidity, radioactivity) associated with DBAs. The equipment is shielded from missiles, pipe whip, and fluid jet forces. Furthermore, each isolation feature is mechanically and electrically redundant to meet the single failure criterion and--for valves--of a fail-closed type. Typically, in the case of a piping penetration, two isolation valves are used. One is located close to the containment wall on the inside and the other is close-coupled on the outside. Figure 7-7 shows some of the valve configurations that meet the design criteria. The variations shown are a function of the termination and origination point of the line of interest, its size, function, and pressure rating.

The isolation system may include a penetration pressurization system in addition to the isolation valves. Pressurization gas--nitrogen--is used to charge a closed volume created by redundant seals in the sleeve cavities on some of the containment piping and electrical penetrations. The presence of the pressurizing gas helps ensure that releases from a pressurized containment are low following a LOCA. The cavity pressure is maintained at a level above that expected to occur inside the containment. Monitoring the pressure during normal plant operation provides an indication of the leak-tightness of the sleeve.

Containment isolation is initiated automatically when excess containment pressure and in some plants overtemperature is

TABLE 7-3. DRY CONTAINMENT PIPING PENETRATIONS (TYPICAL PLANT)

Penetration Function	Number
Chemical and volume control	16
Chilled water	4
Component cooling	7
Containment spray	4
Service water	4
Fire protection	2
Drains to radiation waste system	6
Residual heat removal (RHR)	6
Safety injection	30
Service air	2
Spent fuel cleaning	4
Station heating	2
Instrument air	2
Main steam	36
Process sampling	8
Reactor coolant pressurizer	6
Steam generator blowdown	12
Steam generator feedwater	16
Waste disposal	2
Safety injection	10
Station heating	2
Makeup demineralizer	2
Off-gas	2
Containment purge	8
Instruments	10



Note: M = Manual
MO = Motor Operated
AO = Air operated
AO/S = Air operated with solenoid

Figure 7-7. Isolation valve schemes.
Source: Spencer 1982

detected. All the isolation valves are closed on lines passing through the containment that are not used for the operation of Engineered Safety Features. Some of these lines may be subsequently isolated as well if certain ECCS or containment heat removal systems are activated (indicating an accident of increasing severity).

7.1.5 Long-Term Containment Heat Removal

Nuclear plants utilizing dry containments generally employ two separate long-term containment heat removal systems. Each depends on a different cooling principle. The two systems are the containment water sprays and the atmospheric fan coolers. Both are Engineered Safety Features and are designed to operate after a LOCA despite any single component failure. Thermal energy from the containment is conveyed to the plant's ultimate heat sink. Either system is sized so that it can independently maintain containment pressure and temperature below the appropriate design limits for accidents in the DBA set. Each is powered by electrically driven water circulation pumps. The extent to which each is divided into multiple redundant subsystems varies from plant-to-plant as does the capacity of the individual systems. Typically, the systems are sized to accommodate energies associated with metal/water reactions, sensible and latent heat of the primary system coolant, and reactor decay heat.

The spray system initially takes suction from the borated-water refueling pool or sodium hydroxide storage tank and, subsequently, when the initial source is depleted, from the emergency sump in the primary containment. Drain paths are arranged to channel spray water and steam condensate back down to the emergency sump in the containment basemat. Multiple steel pipe spray headers and nozzles are positioned high in the containment dome at a point where they can deluge a large portion of the contained space. The addition of sodium hydroxide to the spray water is intended to improve the radio-iodine scrubbing capability of the spray. Figure 7-8 shows how the major parts of the spray system are interconnected in a representative plant having a dry containment. Not shown in the figure are various interconnections between the spray system and other ECCS and RHR systems.

The spray nozzles break the water flow into small droplets (500 to 1000 microns in size) in order to enhance the cooling effect of the water. The droplets may fall up to 120 ft through the containment atmosphere before they reach thermal equilibrium.

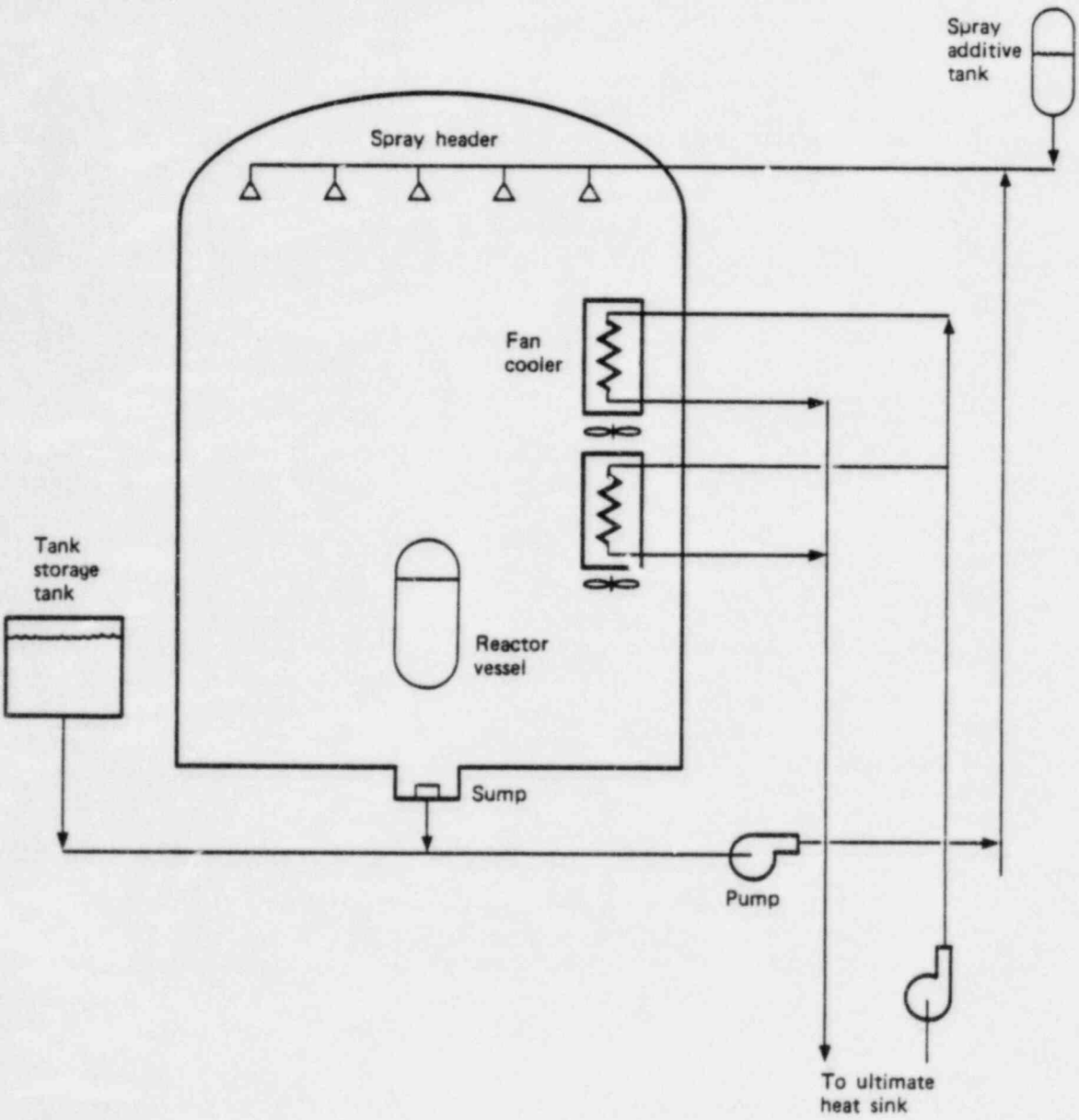


Figure 7-8. Long-term dry containment heat removal (redundant system not shown).

The RHR heat exchangers provide a mechanism for conveying thermal energy in the water drawn from the emergency sump to the plant's ultimate heat sink (e.g., spray pond, ocean, lake, river). Multiple and redundant exchangers are used for this purpose. The cooling water circulating on the secondary side of the units requires electrical power for operation as does the spray water circulation. The design capacity of all the exchangers ranges up to 300,000,000 Btu/hr. Since all aspects of the RHR qualify as Engineered Safety Features, the failure of a single exchanger for whatever reason does not, in itself, prevent adequate heat removal under worst-case LOCA conditions. Most plants have from two to four RHR heat exchangers located within the primary containment.

Containment fan coolers are the second type of Engineered Safety Feature designed for long-term containment heat removal (see Figure 7-8). Cooling water is pumped to the finned tube cooling units suspended in the containment from the ultimate heat sink (with or without an intermediate heat exchanger). Axial flow fans circulate air from the containment atmosphere through the multiple cooling coils in a ducting arrangement designed so that failure of one unit does not adversely affect the others. The fan cooler systems are able to withstand the temperature, pressure, and humidity conditions created by a LOCA. During a LOCA, the fans provide both air cooling and steam condensation functions. Condensate flows to the containment emergency sump for use by the spray or ECCSs. Total cooling water flow to the fan coolers is 6000 gal/min, and the sizing of the individual fan units is such that failure of a single unit (plus inoperation of the containment spray system) does not prevent heat removal at a rate sufficient to protect the containment under DBA conditions. The fan coolers may also have the secondary function of containment heat removal during normal plant operations. Actuation of the containment sprays and fan coolers is triggered by a high containment pressure or the plant operator.

7.1.6 Combustible Gas Control

Combustible gas (hydrogen) control in dry containments is accomplished through the use of (1) an electronic monitoring system that alerts the plant operator to hydrogen concentration at several points in the containment, (2) hydrogen recombiners, (3) containment atmospheric mixing (may be natural circulation induced in some plants), and (4) containment purge to the outside environment. Each of these elements is an Engineered Safety Feature and is designed to function during normal plant operation as well as during DBAs. Redundancy is built in and the subsystems can operate

successfully even in the event of single component failures and loss of off-site electrical power (subsystems requiring electrical power are tied to emergency buses). The design objective is to maintain combustible gas concentrations to less than 4 percent by volume as required in Regulatory Guide 1.7. The purge system serves as a backup to the recombiners in meeting this design objective.

Hydrogen may be released to the containment in appreciable quantities if the accident has progressed to the degraded core stage. Several thousand pounds of hydrogen may result from metal/water reactions, corrosion, and radiolytic decomposition of water. The accumulation of the gas presents a threat to containment integrity if a sizeable quantity were present and if it were to burn or detonate.

The recombiner processes gas drawn from the containment atmosphere at a rate of 50 to 100 cfm. This stream could contain as much as 5 percent hydrogen (for the design basis) with the balance consisting of oxygen, nitrogen and water vapor. The gas stream is heated electrically to the point where the hydrogen/oxygen reaction will occur spontaneously. The hot exit gases from the exothermic reaction are cooled with a forced stream of containment air before return. Each of the redundant recombiners must be activated manually by the plant operator.

Hydrogen/air mixing minimizes the accumulation of hydrogen pockets of dangerously high concentration. Mixing is achieved by natural circulation, fan cooler operation, and/or containment spray operation. Both the fan coolers and the spray system are Engineered Safety Features and are likely to be activated during LOCAs which may release hydrogen to the containment.

The containment purge system is designed as a backup to the hydrogen recombiners. It can be used only if the activity level is sufficiently low to allow venting of the containment atmosphere to the outside environment. Purge air with any contained hydrogen passes through a prefilter, the HEPA filter, the charcoal filter, and the plant stack. Clean fresh air can be added to the containment for make-up purposes. The purge system is controlled manually and is designed to handle about 400 cfm.

7.2 DOMINANT FAILURE MODES OF DRY CONTAINMENT

The dominant failure modes in severe accidents for PWR dry containments have been studied in a number of PRAs beginning with RSS (NRC, 1975) and more recently with the Zion and Indian Point studies (Commonwealth Edison Company, 1981;

1981; Consolidated Edison Company, 1982) and reviews (NRC, 1981; Pratt, 1982; BNL, 1983). The most logical accident sequences leading to core melt for these containments have also been studied and have been catalogued in a number of reports (Kolaczkowski, 1983; IDCOR, 1982; Joksimovich et al., 1983). This section summarizes these results and presents both the dominant accident sequences and the dominant containment failure modes.

7.2.1 Dominant Accident Sequences

The Accident Sequence Evaluation Program (ASEP) draft report (Kolaczkowski, 1983) provides valuable insight into the relative significance of the various accident sequences for PWRs. The program finds it convenient to group the various accidents into sixteen sequences which are shown in Table 7-4. The definitions are based upon taking the dominant accident sequences as defined in each PRA and grouping the sequences first by initiating event type (transient or LOCA) and then by major functional failures. These sequences can be grouped as follows:

a. Transients with loss of the core cooling function (1,3,4).

b. Transients with the loss of the core cooling function and containment heat removal function (station blackout) (2).

c. Transients with the loss of the reactor subcriticality function (ATWS) (5).

d. Transients with an induced loss of reactor coolant system (RCS) integrity and the loss of the core cooling function (6,7,8,9).

e. The "V" sequence, which is failure of a high- to low pressure RCS interface thus causing a potentially nonmitigable LOCA (10).

f. A LOCA with loss of the core cooling function (11,12,13,14).

g. A LOCA with loss of both the core cooling and containment heat removal functions (15,1,6).

After performing a "limited re-baselining" of the PWR results (in an effort at consistency from PRA to PRA), it was determined that sequences 1, 2, 5, 6 and 11 make up the dominant sequences for the PWRs studied (transients with loss of core cooling and with or without containment

TABLE 7-4. PWR ACCIDENT SEQUENCES

Sequence	Sequence Description
1	All types of transients with no core cooling but with containment systems available.
2	TLOOP and TAC/DC bus with no core cooling and no containment cooling.
3	All types of transients with no core cooling and with sprays operable/fans failed.
4	TLOOP and TAC/DC bus with no core cooling and with fans operable/sprays failed.
5	All types of transients with failure to scram (ATWS).
6	Many types of transients with SRV stuck open and no core cooling during injection but with containment systems available.
7	Many types of transients with SRV stuck open and no core cooling during recirculation but with containment systems available.
8	Many types of transients with SRV stuck open and no core cooling during recirculation and with fans operable but sprays failed during recirculation.
9	Spurious safety injection transient with no core cooling during recirculation and with sprays operable/fans failed.
10	"V" sequence (interfacing systems LOCA)
11	All size LOCAs with no core cooling during injection but with containment systems available.
12	Small LOCA with no core cooling during injection but with fans operable/sprays failed.
13	All size LOCAs with no core cooling during recirculation but with containment systems operable.
14	Intermediate small LOCAs with no core cooling during recirculation but with fans operable/sprays failed.
15	Intermediate small LOCAs with no core cooling during recirculation and with no containment cooling during recirculation.
16	Small LOCA with eventual core cooling loss and with no containment cooling caused by the initial loss of containment spray.

Source: Kolaczkowski, 1983.

Key: TLOOP (Transient, Loss of Offsite Power)
TAC AC/DC (Transient, Loss of AC and DC Power)
LOCA (Loss of Coolant Accident)

cooling, and LOCAs with no core cooling during injection, but with containment cooling). For each of the dominant sequences, the following trends were observed:

1. These transients with no core cooling, but with containment cooling, are dependent upon the number of trains for AFWS and HPIS, and possible operator failure to initiate "feed and bleed" where possible. Hence, its dominance is plant-specific.

2. This transient with no core cooling and no containment cooling is driven by station blackout, i.e., loss of all off-site and on-site power.

5. These are ATWS sequences, and their importance depends on "our present understanding of the ATWS scenario for PWRs." In light of the recent event at the Salem plant, the importance of ATWS is uncertain (Raymond, 1983).

6. This sequence is a function of the demand placed on primary system relief valves, particularly the PORV demand rate. Operator action is also an important factor.

11. This class is driven by small LOCAs with failure of emergency coolant injection. Reactor coolant pump seal failures have been the dominant cause of the smaller LOCAs in this class.

13. Although not listed as dominant sequences, LOCAs with no core cooling during recirculation but with containment systems operable may also prove to be dominant at specific plants, and require further study.

The IDCOR program has given the relative weight of initiating events for four PWRs; these are shown in Table 7-5. In examining Table 7-5, two important points should be made. First, the dominance of the sequences described in this section is based on frequency only. Second, only eight internal initiators have been considered. With respect to both points, the relative importance (dominance) of the sequences may change when containment failure (and hence risk) is taken into account. The recent paper by Joksimovich, Frank and Worledge (Joksimovich et al., 1983) highlights this point as shown in Table 7-6.

Generally speaking, the sequences listed in Table 7-6 which lead to early containment failure have a high ranking with

TABLE 7-5. PWR INITIATOR CONTRIBUTIONS TO CORE MELT

Reactor (Type)	Large LOCA[1] (%)	Small LOCA[2] (%)	Transient (%)	External Events[3] (%)
Surry (PWR)	10	70	20	--
Sequoyah (PWR)[4]	10	80	10	--
Oconee (PWR)	5	25	70	--
Zion (PWR)[5]	15	50	30	5

Source: IDCOR, 1982.

1. Includes V sequence
2. Transient-induced LOCAs are grouped with transients
3. Fires, earthquakes, tornadoes
4. Ice-condenser containment
5. From Zion PRA

respect to a significant release. To obtain a ranking based on risk, the product of frequency (core-melt) times the conditional probability of containment failure times the consequence becomes important. The next subsection discusses containment failure modes.

7.2.2 PWR Dry Containment Failure Modes

The principal failure modes for PWR dry containments are summarized in Table 7-7 and can be described as follows:

α: Steam Explosions

In the RSS the steam explosion was a dominant containment failure mode for large LOCAs and some transients. Subsequent work (Theofanous and Saito, 1981; Corridini and Swenson, 1981) indicates a much smaller conditional probability for steam explosions (i.e., less than 10^{-4}), rendering it a small contributor to risk. Recent work by Berman et al. (Berman, 1983) casts some doubt on the 10^{-4} value. Although the actual probability may be unsettled, it is thought to be lower than the earlier RSS value.

β: Failure of Openings and Penetrations

This is potentially an important category, although it is usually allocated a low conditional probability in most PWR PRAs. Its omission from consideration in this report should not be interpreted as a low contributor to risk. The

TABLE 7-6. COMPARISON OF CORE-MELT AND RELEASE SEQUENCES

ZION		
Sequence	Core-Melt Ranking	Significant Release Ranking
Small LOCA: LTDHR failure	1	4
Seismic: AC power loss	2	1
AC power loss and AFWS failure	13	2
Interfacing LOCA	16	3
INDIAN POINT-2		
Seismic: Direct containment failure	21	1
Interfacing LOCA	24	2
INDIAN POINT-3		
High winds: LOOP and SW	11	3
Interfacing LOCA	15	5
Loss of AC power	16	4
Fire in switchgear froom	2	1

Source: Joksimovich, Frank and Worledge, 1983.

TABLE 7-7. PWR DRY CONTAINMENT FAILURE MODES

Mode Description	Description
α	Missiles generated by a steam explosion.
β	Failure as a result of inadequate isolation of openings and penetrations.
γ	Hydrogen burn
δ_1	Overpressure, early in the sequence, and mainly steam.
δ_2	Overpressure, late in the sequence (steam and/or noncondensible gas).
ε	Basemat penetration and melt-through
V	Containment by-pass, e.g., failure of check valves in system piping.

Source: NRC 1975 and NRC 1983.

subject is currently under NRC staff review and its importance will be considered in Task 3 of this project, when specific design and operational features are considered.

γ: Hydrogen Burns

Hydrogen burns become an important contributor to failure if the containment is not steam-inerted and the amount generated during the core meltdown phase is large. With respect to steam inerting, there is a tradeoff between those sequences in which there are containment heat removal systems available, and those where there are not. With containment heat removal, there is little steam and hence a hydrogen burn is important. Without containment heat removal, the steam tends to inert the containment, preventing a burn; failure does however occur due to overpressurization. It should also be noted that some phenomenological uncertainties exist with respect to nonuniform concentrations of hydrogen in containment, as well as to flame acceleration.

δ_1: Early Overpressure

In the RSS, containment failure due to overpressurization was the result of steam evolution (of the primary coolant) at the time of vessel failure. This steam spike was sufficient to cause failure of the Surry containment if it had not failed due to a missile generated by a steam explosion.

δ_2: Late Overpressure

For those sequences without containment heat removal, the containment is assumed to ultimately fail by slow pressurization due to steam energy addition and the generation of noncondensible gases. These noncondensible gases may evolve from core/concrete interactions as well as the hydrogen that does not burn. The late overpressurization failure mode has become dominant because (1) the ultimate capacity of PWR containments has been calculated to be greater than that assumed in the RSS, and (2) recalculations of the peak pressures generated at vessel failure indicate that they are well below these new failure pressures.

ϵ: Basemat Penetration and Melt-Through

In the RSS, it was assumed that if the containment did not fail by any of the above mechanisms, the core would penerate the basemat and eventually melt through it. This melt-through "exhausted" the containment failure modes; i.e., the sum of the conditional probabilities of all the modes summed to one. In the Zion PRA, it was determined that for some sequences the reactor cavity would be flooded with water and hence a coolable debris bed would be formed, thereby preventing basemat penetration. In the absence of water, it was assumed that a dry cavity leads to failure by late overpressurization long before basemat penetration. It should be noted that other assessments (NRC, 1981) indicate that even with a flooded cavity, supplied continuously with water, debris bed coolability is not guaranteed. Moreover, the steam generated with a continuously flooded cavity would ultimately fail containment by the δ_2 mode if containment heat removal fails.

V: Interfacing System LOCA

According to the RSS, the V sequence continues to be an important failure mode for containment, contributing to over 75 percent of the risk. In the Zion PRA, it also dominates risk (in the absence of external initiators). For the purposes of this review, the V sequence can be considered nonmitigable; that is, the check valve failures that lead to the sequence can be eliminated (or their frequency reduced) by prevention. Indeed, operational procedures with regard to inspection and maintenance have been suggested for reducing the effects of the V sequence, but the benefit is doubtful. Uncertainties remain about the relationship between valve leakage and seat failure or pipe leakage before rupture. The desirable frequencies of inspection or monitoring of leakage are unknown.

An example of the conditional probabilities of the various containment failure modes as a function of accident sequence is given in Table 7-8 for the Indian Point Reactors (NRC, 1983). These results, typical of most large dry containments, show the importance of containment heat removal in mitigation systems, and the importance of protecting against basemat penetration.

TABLE 7-8. CONDITIONAL PROBABILITIES OF CONTAINMENT FAILURE - INDIAN POINT REACTORS

Failure Mode	Accident Sequence			
	E	EFC	LF	EF
α	0.0001	0.0001	0.0001	0.0001
β	--	--	--	--
γ	--	0.0001	0.01	0.1000
δ_1	0.005	--	--	--
δ_2	0.40393	0.03069	0.0013	--
ε	0.53187	0.17373	0.1000	0.795
no failure	0.05910	0.79538	0.8890	0.100
	1.0000	1.000	1.000	1.000

Source: NRC, 1983.

E = Early core-melt sequences with no CHRS.
EFC = Early core-melt sequences with CHRS.
LF = Late core-melt sequences with only containment fan coolers operating.
EF = Early core-melt sequences with only containment fan coolers operating.
CHRS = Containment fan coolers and containment spray systems.

7.3 BENEFITS FROM MITIGATION SYSTEMS INSTALLATIONS

The subject of mitigation of severe accidents in large dry PWR containments has been investigated for some time (Gossett, 1977; Carlson, 1978; and Cybulkis, 1978). Recently, there has been a focus on accident mitigation for the Zion and Indian Point nuclear power plants (Murfin, 1980; Ahmad et al., 1983; Kastenberg and Catton, 1983; Gazzillo and Kastenberg, 1983; NRC, 1983). In this section, the results of these later studies are used to illustrate the potential range of benefits achievable in PWR dry containment systems.

7.3.1 Risk Averted

The determination of potential benefits in terms of risk averted for PWR large dry containments will be plant-specific and will depend on the dominant accident sequence frequencies leading to core melt, the conditional probabilities of containment failure (which is a strong function of the containment safeguard systems), and the consequences (which will be site-dependent). As an example of the potential benefits in terms of risk averted, the BNL review of the Zion PRA will be used (Pratt, 1982). This choice is made primarily because the Zion PRA is well documented (and well reviewed), includes external events, and is near the large, high-density population of Chicago. Hence, the benefits achievable should be maximized for this site.

The Zion plant has two containment safeguard systems--sprays and fan coolers. These two systems make up the containment heat removal system. In the BNL review it is convenient to group the dominant accident sequences into six containment classes as shown in Table 7-9. As discussed in Section 7.2, the presence (or nonpresence) of containment heat removal is important in determining the conditional probabilities of containment failure.

As detailed in Section 2.2, the risk averted is determined from the dominant sequence frequencies, the conditional probability of the containment failure mode, and the consequences. These are shown in Tables 7-10, 7-11 and 7-12, respectively, for the BNL suggested values (Pratt, 1982). From these tables, the contribution to risk for each containment failure mode can be obtained. This is shown in Table 7-13. Examination of Table 7-13 indicates that the greatest potential for risk reduction lies in the elimination of the slow overpressurization failure mode, δ_2. It should be noted, however, that there is not general agreement as to the conditional probability for failure due to hydrogen burns in the Class 4 sequences (i.e., early core-melt sequences with the containment sprays operable). There is some evidence to believe that the amount generated in these sequences may be greater.

7.3.2 Value to Mitigation

The potential for accident mitigation in terms of dollar expenditure can be estimated using a suitable algorithm such as the $1000/man-rem averted that was discussed in Section 2.2 of this report. This is shown in Table 7-14 as a function of the importance of the hydrogen burn. These results can be interpreted as follows. A mitigation system designed to eliminate the slow overpressurization failure mode should

TABLE 7-9. BNL SUGGESTED GROUPING OF ACCIDENT SEQUENCES INTO CONTAINMENT CLASSES FOR THE ZION PLANT

BNL Suggested Containment Class	Comments	Sequence Grouping
1	Interfacing system LOCA	V
2	Sequences with no CHRS operation	SE, SL, TE, AE, AL
3	Sequences with failure of ECCI and with fan coolers operating but no spray systems (failure of ECCI implies early core melt).	SEF, TEF, AEF
4	Sequences with failure of ECCI but with spray systems operating (fans may or may not be operating, implies early core melt).	SEFC, SEC, TEFC, TEC, AEFC, AEC
5	Sequences with failure of ECCR but with CHRS operating (failure of ECCR implies late core melt).	SLFC, SLF, SLC, ALFC, ALF, ALC
6	Sequences with no CHRS operating	SE, TEC

Source: Pratt, 1982

Key:

CHRS - Containment heat removal system (sprays and fan coolers)
ECCI - Emergency core coolant injection
ECCR - Emergency core cooling recirculation
A - large LOCAs
S - small LOCAs
T - transients
E - early core melt
L - late core melt
F - fan coolers operating
C - containment sprays operating

TABLE 7-10. SNL SUGGESTED ACCIDENT SEQUENCE FREQUENCIES FOR THE ZION PLANT

Containment Class[a]	Category	Internal Only	External Plus Internal[a]
1	V sequence	1.0×10^{-7}	1.0×10^{-7}
2	No CHRS - internal only	5.6×10^{-6}	5.6×10^{-6}
3	Early core melt - no sprays	3.1×10^{-9}	7.3×10^{-9}
4	Early core melt - with sprays	3.9×10^{-4}	3.9×10^{-4}
5	Late core melt - no sprays	2.7×10^{-5}	2.7×10^{-5}
6	No CHRS - external only	-	5.3×10^{-6}
Total		4.21×10^{-4}	4.26×10^{-4}

Source: Pratt, 1982

[a]Does not include seismic failure of containment.

TABLE 7-11. BNL SUGGESTED CONDITIONAL PROBABILITIES OF CONTAINMENT FAILURE FOR THE ZION PLANT

Containment Class	Failure Mode								
	α	β	γ	δ_1	δ_2	ε	V	External	None
1							1.0		
2					1.0				
3			0.40						0.60
4			0.02						0.98
5			0.01						0.99
6								1.0	

Source: Pratt, 1982

TABLE 7-12. BNL SUGGESTED MEAN CONSEQUENCES FOR EACH CONTAINMENT FAILURE MODE FOR THE ZION PLANT

Failure Mode	Acute Fatalities	Latent Fatalities	Man-Rem
γ	1.1×10^{-3}	525	9.96×10^{6}
δ_2	0	1440	2.41×10^{7}
ν	622	1660	2.8×10^{7}
External - no CHRS	120	1690	2.71×10^{7}

Source: Pratt, 1982.

Basis: RSS source term and methodology.

TABLE 7-13. CONTRIBUTION TO RISK FOR EACH CONTAINMENT FAILURE MODE

Failure Mode	Acute Fatalities/Yr	Latent Fatalities/Yr	Man-Rem/Yr
γ	~0	3.1×10^{-4}	80
δ_2	~0	8.0×10^{-3}	135
ν	6.2×10^{-5}	1.7×10^{-4}	2.8
External - no CHRS	6.4×10^{-4}	9.0×10^{-3}	144
Total	7.0×10^{-4}	17.5×10^{-3}	362

Basis: RSS source term and methodology.

TABLE 7-14. VALUE OF MITIGATION BASED ON $1000/MAN-REM AVERTED (40-YEAR PLANT LIFE, PERFECT MITIGATION)

Late overpressure internal events only (δ_2)	$ 5.4 million
Late overpressure external events only (δ_2)	$ 5.7 million
Hydrogen burn internal + external Cp = 0.02	$ 3.2 million
Total	$14.3 million

Cp = conditional probability of containment failure due to hydrogen.

Basis: RSS source term and methodology.

cost less than $5.4 million if only internal initiators are considered. When external events are included (e.g., fires and seismic events), the "allowable cost" goes to $11 million. However, the mitigation system must be designed to cope with the external event itself. At the Zion plant for example, it was estimated that in 96.6 percent of the earthquake-induced core melts the containment is intact. Hence, the mitigation system must also survive to be of use. For the 3.4 percent of the cases where the containment fails (in the seismic event), the mitigation system would be of no use. At Zion moreover, seismic events resulted in loss of off- and on-site power and, hence, the containment heat removal system. Therefore, any mitigation system that is to survive a seismic event should have its own dedicated power, or be passive.

7.4 MITIGATION OPPORTUNITIES FOR DRY CONTAINMENT

The dry containment has been shown in the previous sections to be subject of substantial residual risk of failure during a severe accident. The risk stems primarily from loss of site power and containment heat removal, leading to slow overpressure failure, and from hydrogen fires at any stage of an accident. Both internal and external events contribute to the accident risk. Listed below are the expected accident conditions that would form the environment for any mitigation effort.

7.4.1 Cumulative List of Accident Conditions

1. All site electrical power is lost and all instruments are inoperative.

2. The emergency core cooling pumps and containment sprays are inoperative.

3. The core has melted, will soon have penetrated the reactor vessel, and will fall into the instrument tube sump.

4. The sump and the floor of the containment may be either dry or covered with water, though the latter is most likely.

5. Much of the Zircaloy in the fuel cladding will have reacted to form hydrogen gas, now released to the containment. The atmosphere of the containment will be saturated with steam.

6. As much as 50 tons of molten steel will accompany the core onto the floor of the containment.

7.4.2 Criteria for Mitigation

The philosophy adopted for design of mitigation systems will be to ensure as positive and defensible a result as possible, which establishes the following further requirements and conditions:

1. Whenever the accident end-state conditions are uncertain and the technical community is not in agreement as to the outcome, the anomalous situation will be circumvented by a design that either avoids or minimizes these uncertainties--the result is a known end-state.

2. Completely passive mitigation systems will be used whenever possible. Where this cannot be done, or is unreasonably costly, a quasi-passive system requiring no personal attention will be used. A fully independent and dedicated source of energy will be used for the task, preferably with installation in duplicate.

3. Controlled venting of the containment is always preferable to any form of uncontrolled failure of the structure or the penetrations. An intact containment always presents less risk to the area than a ruptured one. Where venting is necessary, more than one valve will be used, of the self-reclosing type that will release only the minimum

amount of gas or vapor necessary to keep the containment pressure within safe limits.

4. Maintaining directional control and adequate cooling of the debris from a molten core at all times will present less risk and less future cost than allowing any of it to escape into the environment or to indefinite locations within the containment, where its subsequent behavior is uncertain.

5. Mitigation equipment and its operation must present minimal interference to the normal operation of the plant.

6. Mitigation equipment should be of safety grade quality, but need not be documented as such. Where the system penetrates the containment, the usual regulatory requirements must be satisfied.

7. Where possible, alternative means of handling these accident conditions will be considered. Carefully defined operator actions could be such an alternative.

7.4.3 Potential Accident End-States to Be Considered

The following situations may exist when mitigation is undertaken:

1. The containment is filled with a mixture of air, steam, and hydrogen. The atmosphere is either flammable, near-flammable or steam-inerted. Ignition of the hydrogen would probably fail the conment, for example, if power were restored and the sprays came on.

2. The molten core is generating more steam at about 100,000 lb/hr, and may be generating other gases from concrete attack as well. The containment pressure is increasing toward the failure point.

7.4.4 Cumulative List of Mitigation Functions to be Performed

The mitigation system must accomplish the following functions:

1. The possibility of a hydrogen burn must be removed by some means, and the hydrogen must be disposed of.

2. Heat removal from the containment without release of radioactive materials must be sufficient to prevent overpressure from steam generation.

3. Attack of the concrete basemat must be prevented to limit the formation of noncondensible gases.

4. As heat is removed and steam condenses, underpressure failure of the containment must be prevented.

7.4.5 Candidate Means for Meeting Requirements

Mitigation techniques and devices known in the literature that are potentially suitable for meeting the above requirements have been discussed in Hammond (1982), Swanson (1983), Murfin (1980), and Ahmad (1983), and are reviewed briefly in the Task 2 report. The mitigation systems considered include filtered vent systems, hydrogen control systems, containment heat removal systems, and core retention systems.

The slow rise in pressure from the heating effects of the escaped core debris represents two kinds of gas generation: steam from any water present in the reactor compartment sump; and steam, carbon dioxide, and other gases from thermal decomposition of the concrete floor. Mitigation systems for these effects would consist of providing a prepared location below the reactor vessel into which the core debris can be confined and cooled. In installations where a core retention device is retrofitted into an existing plant, either a prepared bed in the existing instrument-tube sump (the thoria pebble bed (Swanson, 1983) or a water-cooled crucible (the crucible in a tunnel (Hammond, 1982) might be used. Backfitting a receptacle could be somewhat expensive because of high radiation levels, necessitating a remote control installation.

Maintaining the quenched condition requires a flow of cooled water over the core debris or over the walls of the crucible container. In principle, such water circulation and cooling could be accomplished by natural circulation, but dedicated diesel engines driving pumps could also be used and may be more effective. Well-known industrial equipment is available for this function at a reasonable cost. Heat pipes, large single condenser systems, and water sprays can also be used to provide containment heat removal. Heat pipe systems have been studied for this purpose in some detail (Ahmad, 1983).

Hydrogen control systems proposed for use in large dry containments include spark ignition, glow plugs, open flames, water sprays, fog nozzles, and foams (Murfin, 1980; Levy, 1981). The only fully recognized successful method of doing this task is operation of the plant with an inert atmosphere always in place. Since this would make very difficult and expensive any preventive maintenance in the containment during operation, several schemes for rapid depletion of the oxygen concentration upon demand have been considered (see the Task 2 report). Filling the containment with fire-fighting foam, fog, or one of the Freon gases could all be done rapidly, and such methods have been used successfully in industry. Another possibility is to consume the oxygen by burning a fuel, and removing the resulting heat. To accomplish this with a reasonable size of burner and cooling system would mean beginning the inerting process early in the accident, perhaps before it was clear that a core melt would actually occur.

In new construction, there are several other core retention systems that could be considered for use. A suitably large vacuum release valve will be required to prevent inward collapse of a steam-filled containment when cooling exceeds the heat generation rate. When added heat removal capacity is brought in, this problem becomes critical. This release can be accomplished with standard industrial valves designed for this purpose. Perhaps two valves in parallel may be necessary to meet redundancy requirements. The cost will be low.

All of the systems mentioned above appear to be feasible in the large dry type of containment. They can generally be retrofitted into existing systems and should not interfere with the normal operation of the reactor. Qualifications concerning their use are discussed in greater detail in the accompanying Task 2 report.

7.5 SUMMARY

The PWR plants having large dry containment vessels represent the largest class of nuclear plants. The large capacity of the containment makes it tolerant of many of the accident sequences that cause early failure of other types. Thus, few of the dominant failure modes of large containments occur quickly, so that there is more time to take remedial action compared to other containments. The design of mitigation systems should be made easier and thereby more versatile.

The largest residual risk for initial failure comes from slow overpressurization caused by loss of electric power and containment heat removal, in most cases accompanied by con-

crete attack. Whether the initiating event is from internal or external causes, a core-melt accident is almost certainly accompanied by hydrogen formation during the dryout of the core. The large volume of the containment means that this hydrogen does not in itself cause overpressurization, even combined with steam and other gases in the early part of the accident.

There is no doubt, however, that a flammable or explosive mixture is likely to be formed at some stage in every severe accident, possibly quite early after primary system rupture. The energy of combustion released if burning does initiate is clearly more than enough to rupture the containment. Deliberate ignition could be disastrous unless precise knowledge is available that only a small amount of hydrogen is present and that heat removal systems can keep up with the burn rate or the evolution rate. The uncertainty of whether a flammable mixture will persist without self-ignition does not seem to be a tolerable one within the ground rules set up for this study, so some mitigation method is essential for this condition. At present, this requirement is farthest from having a clearly proven solution, and further study is recommended.

The large dry containment system and its failure modes have been subject to several reviews. These reviews have tended to bring the PRA results to a level of consistency, permitting some conclusions about absolute levels of risk that could be averted by mitigation. As shown in Table 7-14, the algorithm of $1000 per man-rem averted leads to dollar levels of $3 to $11 million for the individual failure modes (overpressure plus hydrogen burn). Because of the uncertainty in the importance of the hydrogen burn, the upper bound can be greater. For example if $C_p = 0.2$, the value of hydrogen mitigation becomes $32 million. Preliminary indications are that a complete mitigation system should be obtainable within the total sum ($15 to $32 million). Table 7-15 shows risk reduction factors based on acute fatalities, latent fatalities, and man-rem.

TABLE 7-15. RISK REDUCTION FACTORS

$$F = \frac{\text{Risk without mitigation (Rw/o)}}{\text{Risk with mitigation (Rw)}}$$

Failure Mode	Acute Fatalities/Yr	Latent Fatalities/Yr	Man-Rem/Yr
Y	1.0	1.017	1.28
O_2	1.0	1.84	1.59
V	1.09	1.0098	1.006
External - no CHRS	128.00	2.061	1.66
All overpressure, + Ext.	128.00	35.00	4.36

CHAPTER 8. SUMMARY AND CONCLUSIONS

This report provides an analysis and assessment of the five major types of reactor containment from the standpoint of severe accident mitigation. Each chapter describes one of the five principal containment types to show how they are intended to function under DBA conditions, how they might fail under severe accident conditions (i.e., beyond their design basis), and what changes might be feasible to increase their capability of withstanding such accidents.

The intent of this report, the first in a series of five, is to provide an assessment of the opportunities for severe accident mitigation and where they are likely to be most beneficial. In addition, a convenient data base is provided where, for the first time, comparative information on the major types of containment is available in the same document. This scoping study should be regarded as giving direction, rather than conclusions, since its findings are necessarily qualitative in nature. Final conclusions and formal decisionmaking should be based on detailed design and cost determinations.

To place this report in perspective, the objective for the overall project can be briefly stated as that of finding answers for five questions:

1. What is the present state of containment design and function, and the present understanding of containment behavior, with respect to severe accidents?

2. What is the present state of the art of severe accident mitigation technology?

3. Is it technically feasible to apply this technology to existing and new containments, making them highly resistant to failure under severe accident conditions?

4. Would such changes reduce the residual risk to the public to a degree sufficient to justify the cost involved?

5. How might a regulatory policy focused on mitigation be developed and implemented?

This report addresses the first question and provides preliminary answers for the second and third. For the fourth question, the report addresses the state of the art

of risk measurement and the potential for risk reduction, and reports the meager cost information in the literature. No new cost information is developed. The fifth question will be addressed in a later report.

The findings of this report represent the opinions of the authors only, and do not necessarily reflect policies or opinions of the NRC or other agencies.

8.1 ARRANGEMENT AND FUNCTION OF REACTOR CONTAINMENTS

The description of U.S. reactors and their containments presented here was undertaken to provide a basis for relating their arrangement and function to the phenomena imposed by a severe accident. This information is a vital part of the conclusions reached in this report and those that follow. However, it was not a part of this study to assess or compare these containment systems as to their capability in meeting normal operating conditions and withstanding DBAs--a subject covered thoroughly in the licensing process.

The safety systems of these reactor types are described briefly, however, in order that potential interactions with mitigation activities could be considered. Also presented are design and performance data such as containment volume and pressure rating needed for determining overpressure conditions. The types of containments studied represent three generations of design for the BWRs, and at least two for the PWRs. The later designs, representing larger thermal capacities, have larger volumes and greater overall capabilities, but they are still based on meeting specific DBA conditions.

The main conclusion with respect to the overall structure and function of the five major containment types can be simply stated: Conceptually, it appears to be structurally feasible to modify each type of containment to survive or mitigate a major core-melt accident. Without such modification, all would have a significant probability of failure in a severe accident.

8.2 COMPARISON OF THE FAILURE MODES OF DIFFERENT CONTAINMENTS

Under severe accident conditions, the five major containment types reveal a similar degree of vulnerability to one mode of failure, namely overpressurization caused by slow accumulation of noncondensible gases or steam, with failure of containment heat removal systems. This condition forms a significant part of the failure probability for each containment type. For a large dry containment, the vulnerability comes from loss of the containment spray systems or

the fan coolers. For BWRs, the vulnerability comes from the loss of suppression pool cooling, as well as the loss of containment sprays. It is important to note that these systems (fan coolers, sprays, suppression pools) were designed to cope with DBAs, yet their active presence during a core-melt accident can serve to delay or prevent containment failure. Their loss is usually attributed to loss of all electrical power, common mode failures, and/or operator error.

Hydrogen formation is a highly variable threat to the different containments. For hydrogen gas formation alone, without burning or concurrent steam formation, none are likely to be significantly overpressured. The Mark I and Mark II containments, having inert atmospheres, are not susceptible to hydrogen conflagration or detonation, but all the rest would be expected to fail if an optimum burn occurred. When hydrogen formation is combined with rapid formation of steam and other gases, all but the large dry and Mark III containments tend to be threatened by a rapid overpressure failure. This situation is especially aggravated for the BWRs in case of an ATWS, where the suppression pool becomes fully saturated before the core melt begins.

8.3 UNCERTAINTIES IN CORE MELT PHENOMENOLOGY

A great deal of uncertainty in our understanding of core melt phenomenology still exists. The initial stages of core melt progression are most important to the development of a mitigation strategy. Several aspects of these uncertainties are as follows:

1. Early Core Degradation Period. Several unpublished studies point to the possibility of early primary system failure. The core region heats the upper plenum by recirculation. The hot gas deposits radionuclicles in the upper plenum and hot legs. It is speculated that this could lead to hot leg nozzle failure or even steam generator tube rupture. This could change the dominant order of the accident sequences.

2. Core Melt Progression. German studies show uniform progression of the melt front with the melt being at relatively low temperatures. SNL measurements of the heat transfer coefficient show that it is large. The melt temperature may be much lower than heretofore believed. This lowers the in-vessel decontamination factor and could lead to more radioactivity in the containment

building. One of the most important variables in source term calculations is melt temperature, and it has not been calculated adequately at present.

3. Molten Material Ejection Process. Sequences leading to high pressure at the time of vessel failure are analyzed in the manner described in the Zion PRA (BNL, 1983). It is postulated that a liquid jet, choked at the vessel breach, will exist until all the molten material is ejected. This is a very fast process. The result is a possibility that all the thermal and chemical energy may be added to the containment atmosphere so quickly that overpressurization occurs. This is primarily a problem for PWRs. However, there is reason to believe that the above scenario will not occur. Rather, high-pressure gas will be blown out the breach and the emptying process will be significantly prolonged. The hold-up process will allow additional failures and multiple breaches to develop. This tends to speed up the emptying process. The actual rate of ejection is not known, but is very important in the development of mitigation systems.

4. The question of steam explosions is not yet fully resolved, and new evidence seems to point to the core material being more explosive than heretofore thought. Under some circumstances we may be missing the dominant failure mode. Not knowing the melt temperature will lead to difficulties in estimating the source terms. The rapid emptying process that could lead to direct heating and rapid pressure increases cannot be ruled out at this time.

8.4 NATURE AND FEASIBILITY OF STRUCTURAL AND PROCEDURAL MODIFICATIONS TO MITIGATE SEVERE ACCIDENTS

As noted in Chapter 2, the occurrence of a severe core-melt accident affects different containments in different ways; but when operator actions or mitigation devices intervene to prohibit the most probable mode of containment failure, the accident itself does not cease. The core residue continues to produce decay heat, and threatens the containment by producing steam, gas and radiant energy until it either emerges into the environment below the basemat or is arrested in a quenched condition with continuous heat removal.

Although different modes of breaching the containment may have varying degrees of risk to humans and property, we have taken as a premise that any initial investigation of mitigation as a possible regulatory policy must be based on essentially complete containment failure prevention. We believe that to recommend intervention in an accident by preventing one mode of failure, only to ensure that a different one follows, is insupportable unless proof is available that the secondary failure mode carries negligible public risk. Although we agree with the argument that delaying the time of containment failure tends to reduce the risk, no quantitative estimates are available, especially for the present case, wherein a severe accident has been "bottled up" by preventing its normal outcome. Recent work (CSNI, 1983) has shown that thermal-hydraulic processes normally unimportant can assume a significant but as yet unassessed role. Current PRAs, on which this study was necessarily based, certainly do not consider the potential increase in conditional probability of containment failure from one mode when others are prohibited. Although applicable data will appear in time, the only safe assertion initially is that mitigation should be "complete."

With this view, we find that mitigation systems selected for the different containment types tend to require much the same kinds of elements: means to prevent or relieve overpressure, dedicated means of adequate containment heat removal, and means for retaining and cooling the core residue. For the containments that are not already inerted, the threat of a hydrogen burn is real and must be dealt with.

Our study of the state of the art of mitigation systems, the subject of a companion report, will determine where reliable means exist to control phenomena accompanying a severe accident for each type of containment, and to fulfill the requirements for complete mitigation. The information available in the literature on costs of such mitigation systems is too meager to permit any conclusions on this subject. Work presently underway for subsequent reports will provide information on both costs and benefits.

8.5 POTENTIAL BENEFITS OF MITIGATION

In Chapter 2, two expressions for measuring the potential effect of mitigation were discussed: the difference in risk without and with mitigation was defined as the <u>risk averted</u>, and the ratio of the risk without mitigation to the risk with mitigation was defined as a <u>risk reduction factor</u>. In this context, risk is taken as frequency times consequence, as measured for each consequence, e.g., man-rem, early death, latent death, etc. Unfortunately, the

risk values obtained from the different PRAs are based on different assumptions, procedures, scope and detail. This variability, which became apparent during the course of this project, prevents any broad-based conclusions at this time with respect to comparative cost effectiveness.

In an attempt to determine where mitigation opportunities exist, Table 8-1 exhibits risk reduction factors for each containment type. Because of the variability in the scope, detail and review of each PRA, the factors shown in Table 8-1 should be viewed in a qualitative way. They do, however, indicate the following:

1. The potential benefit of mitigation with respect to steam explosions (both in-vessel and ex-vessel) is small. This occurs because the PRAs tend to assume the conditional probability of a steam spike causing containment failure is small.

2. The potential benefit of mitigation against overpressurization due to steam evolution and/or noncondensibles is great (an order of magnitude or more) for all containments except the ice condenser, where the hydrogen threat dominates. In this case, there is a potential risk reduction of 4, if overpressure failure can be eliminated.

3. The potential benefit of mitigation against a hydrogen burn or detonation is large in BWR Mark III containments (which are not inerted as are the Mark I and II) and in the ice condenser (because of a low pressure rating).

For two containment types, the BWR Mark II (based on the Limerick PRA) and the PWR large dry containment (based on the Zion PRA), there has been broadened scope and considerable review of the PRAs and their analyses. In both cases, the PRAs have included external events (e.g., seismic, fires, etc.) and have been reviewed extensively by the National Laboratories.

For these two PRAs, we have employed the NRC's proposed cost effectiveness criterion of $1000/man-rem averted to determine a justifiable level of expenditure for achieving perfect mitigation. These two results are shown in Table 8-2. A 40-year plant life with no monetary discounting was assumed.

In both cases, "engineering judgment" indicates that the amounts shown could be sufficient to mitigate the major

TABLE 8-1. RISK REDUCTION FACTORS BASED ON MAN REM AND EXISTING RRA INFORMATION (RATIO OF RISK WITH CONTAINMENT FAILURE MODE TO WITHOUT)

Mode of Containment Failure	Containment Type				
	BWR Mark I	BWR* Mark II	BWR*** Mark III	PWR* Large Dry	PWR Ice Cond.
Steam explosion (in vessel)	1.0	1.0	1.0	1.0	1.0
Steam explosion (ex-vessel)	1.0	1.0	1.0	1.0	1.0
Early overpressure	---	1.01+	1.05+	---	4.4
Late overpressure	50	10**	33 (160++)	35	---
Hydrogen burn	---	---	---	1.2	13.3
Containment bypass	---	---	1.0	1.1	1.2

*Includes external events.

**Includes large leaks due to overpressurization.

***Based on Grand Gulf PRA.

+Assumes an ATWS fix.

++Value in parenthesis denotes acute fatalities.

TABLE 8-2. ALLOWABLE COSTS BASED ON $1000/MAN-REM AVERTED ASSUMING 40-YEAR PLANT LIFE, NO DISCOUNTING

LIMERICK	
Case	Amount Available ($M)
1. Mitigate overpressure - internal events only	15.7
2. Mitigate overpressure - external events only	4.2
Total available	19.9

ZION	
Case	Amount Available ($M)
1. Mitigate late overpressure - internal events only	5.4
2. Mitigate late overpressure - external events only	5.7
3. Mitigate hydrogen	3.2
Total available	14.3

threats to containment. Further detailed studies in this program are aimed at:

1. Providing detailed design and cost information for mitigation systems in particular plants.

2. Providing detailed information on the benefits in terms of risk averted.

3. Defining appropriate value/impact measures for assessing whether those mitigation systems are cost effective.

It is expected that as the other containment types receive more analysis, assessment, and review, there will be similar mitigation opportunities which have the potential for cost-effective implementation.

8.6 THE APPLICABILITY OF OPERATOR ACTION, UPGRADING OF EXISTING FACILITIES, AND PROCEDURAL CHANGES TO MITIGATION

The emphasis in this study has been upon "hard" mitigation systems--structural and mechanical systems designed to intervene in the accident progression, force it to follow a preferred rather than a random outcome, and relieve at each stage the conditions that could cause containment rupture. It has been assumed that these systems could not rely upon the availability of electric power nor upon operator action for their function. With the information available to us during the study, no other course could have been chosen. However, it is obvious that for a specific plant at a particular site any design for a mitigation system should make maximum use of existing structures and systems, although the extent of this cannot be predicted before the final design stage.

It is also obvious that operator action can play an important role in accident mitigation providing there is enough time. Such a strategy could potentially be much more cost effective than dedicated automatic systems with fail-safe initiating methods. A correcting factor would have to be applied to such a mitigation strategy to take into account the reliability factor of the human action. At present, in view of the controversial state of assigning such human factors, the use of operator action can be concluded to be a real but future possibility.

Last, it is obvious that changes in current operating procedures, both inside the plant (e.g. venting the containment before core melt) and outside (e.g. alternative evacuation schemes) may offer cost-effective reductions in risk. Later

activities in this project will develop value/impact measures which can be used as a basis for comparing these alternatives. Such comparisons can play a part in developing regulatory strategies for mitigating residual risk.

LIST OF ACRONYMS AND ABBREVIATIONS

AC	Alternating Current
ACRS	Advisory Committee on Reactor Safeguards
ADHR	Advanced or Alternate Decay Heat Removal
ADS	Automatic Depressurization System
AEC	Atomic Energy Commission (Predecessor of the NRC)
ALARA	As Low as Reasonably Achievable
AIF	Atomic Industrial Forum
ASEP	Accident Sequence Evaluation Program
ATWS	Anticipated Transient (S) without Scram
AWPLS	Air and Water Positive Leakage Control System
BCL	Battelle Columbus Laboratory
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CCFF	Complementary Cumulative Frequency Function
CHF	Critical Heat Flux
CMA	Core Melt Accident
Corium	Core Debris after a CMA
CP	Construction Permit
CRD	Control Rod Drive
CS	Core Spray
CST	Condensate Storage Tank
CT	Cooling Tower
CVRS	Containment Vacuum Relief System
DBA	Design Basis Accident

DC	Direct Current
DF	Decontamination Factor
DG	Diesel Generator
DHR	Decay Heat Removal
"DOM-INANT"	Pertains to Accident Risk Contribution > 1% of Total Risk
ECCS	Emergency Core Cooling Systems
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
ESW	Essential Service Water
ETA	Event Tree Analysis
FC	Fails Closed
FITS	Fraction of Core Energy Generating Steam Spike
FO	Fails Open
FSAR	Final Safety Analysis Report
FVCS	Filtered Vent Containment System
FTA	Fault Tree Analysis
FW	Feedwater
GE	General Electric Company
GESSAR	GE Standard Safety Analysis Report (Advanced BWR)
HEPA	High Efficiency Particulate Air/Absolute (Referring to the SGTS Filters)
HP	High Pressure
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
H&V	Heating and Ventilating

HVAC	Heating Ventilating and Air Conditioning
HX	Heat Exchanger
IAC	Interim Acceptance Criteria (AEC)
IBV	Inboard Isolation Valve
IDCOR	Industry Degraded Core Rulemaking Program
I&C	Instrumentation and Control
ID	Inside Diameter
IEEE	Institute of Electrical and Electronic Engineers
INEL	Idaho National Engineering Laboratory
IORV	Inadvertent Open Relief Valve
IRM	Intermediate Range Monitor
LC	Locked Closed
LCO	Limiting Condition for Operation
LCS	Leakage Control System
LDS	Leak Detection System
LFMG	Low Frequency Motor Generator
LGS	Limerick Generating Station
LO	Locked Open
LOCA	Loss of Coolant Accident
LOFW	Loss of Feedwater
LOHR	Loss of Heat Removal
LOOP	Loss of Offsite Power
LP	Low Pressure
LPC	Low Pressure Cooling
LPCC	Low Pressure Core Cooling

LPCI	Low Pressure Coolant Injection (a Mode of RHR)
LPCS	Low Pressure Core Spray
LPRM	Local Power Range Monitor
MARCH	Meltdown Accident Response Characteristics
MCC	Motor Control Center
ML	Manufacturing License
MMH	Monorail Mounted Hoist
MOV	Motor Operated Valve
MSIV	Main Stream Isolation Valves
NC	Normally Closed
NLC	Normally Locked Closed
NLO	Normally Locked Open
NO	Normally Open
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
NSSS	Nuclear Steam Supply System
NUREG	NRC Report Number Prefix
OBV	Outboard Isolation Valve
OD	Outside Diameter
OL	Operating License
OPREL	Overpressure Release
ORNL	Oak Ridge National Laboratory
OXRE	Oxidation Release
PRA	Probabilistic Risk Assessment
PECO	Philadelphia Electric Company
PRM	Power Range Monitor

PRV	Pressure Relief Valve
PSAR	Preliminary Safety Analysis Report
PWR	Pressurized Water Reactor
RCI	Reactor Core Isolation
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RDA	R & D Associates
RES	Research Branch of NRC
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSS	Reactor Safety Study (WASH-1400)
RSSMAP	RSS Methodology and Applications Program
RWCU	Reactor Water Clean Up
SACM	Severe Accident Consequences Mitigation
SAR	Safety Analysis Report
SASA	Severe Accident Sequence Analysis
SARRP	Severe Accident Risk Reduction Program
SAUNA	Severe Accident Uncertainty Analysis
SCFM	Standard Cubic Feet Per Minute
SCRAM	Introduction of Control so as to Stop Criticality
SGTS	Standby Gas Treatment System
SNL	Sandia National Laboratories
SORV	Stuck Open Relief Valve (Subsequent to an Initiating Event)

SP	Suppression Pool
SPASM	System Probabilistic Analysis by Sampling Method
S/RV	Safety/Relief Valve
SW	Service Water
TCV	Turbine Control Valve
TG	Turbine Generator
TMI	Three-Mile Island Nuclear Plant
UHS	Ultimate Heat Sink

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APPENDIX A. SAFETY SYSTEM FUNCTIONS IN LIGHT WATER REACTORS (LWRs)

The underlying safety strategy which traditionally serves as a basis for the design, licensing, construction, and operation of commercial LWR power plants in the United States is discussed in this Appendix. The discussion focuses on the ten major safety functions applicable to the plants and the relationship of these functions to severe accident mitigation systems. The safety functions form a complete set which satisfies the requirements of 10 CFR 50 for licensing by the NRC of domestic nuclear power utilization facilities. Both Appendix A of 10 CFR 50 and the NRC Regulatory Guides (published by the NRC Office of Standards Development) provide guidance to the specific safety system capabilities expected and standards against which their performance must be judged.

The objective sought through the incorporation of safety functions and a corresponding strategy for their use is to control the potential release of harmful radioactive materials from the plant. The extent to which some loss of these materials can be tolerated is specified in terms of "as low as reasonably achievable" (ALARA - see 10 CFR 50, Appendix I) and, among other things, in terms of numerical radiation limits on gaseous and liquid effluents. For example, the total quantity of all radioactive material above the normal background released from the plant to the atmosphere in unrestricted areas will not result in an annual dose greater than 10 millirads for gamma radiation and 20 millirads for beta radiation. It is desirable that mitigation systems be able to achieve these same objectives even in the event of core disruptive accidents.

A classification system sometimes used to categorize nuclear plant events is shown in Table A-1. Nine classes are called out, with the severity of individual events ranging from trivial (Class 1) to catastrophic (Class 9). A nuclear plant must currently be able to meet NRC safety criteria for all events except those in Class 9 in order to acquire an operating license. Events up to the next most severe category (Class 8) form the Design Bases Accident (DBA) set. The plant systems designed to cope with DBAs are referred to as Engineered Safety Features. Accident sequences involving core degradation belong to Class 9.

The conventional nuclear plant's safety functions can be divided into four basic classes. These classes pertain to (1) prevention of melt, (2) containment integrity, (3) control of indirect radioactive releases, and (4) maintenance

TABLE A-1. REACTOR ACCIDENT CLASSIFICATION

Class	Description
1	Trivial occurrences
2	Small releases outside the containment
3	Failures in the radwaste system
4	Release of radioactivity into the primary system
5	Release of radioactivity into the secondary system (does not apply to a BWR)
6	Refueling accidents within the containment system
7	Spent fuel accidents outside the containment system
8	Accidents considered in the design basis (e.g., steam piping break, reactivity transient)
9	Failures more severe than Class 8 (e.g., vessel rupture, core melt, massive earthquake damage)

of vital auxiliaries relied upon by other safety functions (Corcoran, 1981). The ten safety functions that constitute the four classes are shown in Table A-2. Historically, it has been thought that the health and safety of the general public was adequately protected if safety functions in all four classes were successful in the event of abnormal plant conditions. Now, however, when considering severe accident mitigation systems, it is assumed that the safety functions in Class 1 (prevention of core melt) have failed to perform as expected and the consequences of the resultant core damage are to be mitigated. The failure of Class 1 functions may be partially a result of failures in Class 4 functions (such as primary and backup electrical power), massive external events, sabotage or other occurrences. Since the safety functions in Class 2 are normally designed under the assumption that core melt does not occur, they too may be overstressed and fail in the event of core degradation. The combination of core degradation and containment integrity failure is likely to result in unacceptable release of radioactive materials. This is the situation that calls for accident mitigation. Severe accident mitigation requires that additional plant provisions be made to maintain containment integrity even when core degradation has occurred. Alternatively, mitigation systems may take some other steps to control the release of radioactive effluents in the absence of containment integrity. The ten basic safety functions are described below. Each of these is applicable to either a BWR or PWR power plant.

Five functions make up the core-melt prevention class. The first, reactivity control, is achievable by control rod insertion, natural voiding within the reactor core, or borated water addition. This multiplicity of possible success paths to achieve the safety goal is common to all functions. The choice of the particular path to be followed depends on the state of the plant and the degree of operator intervention. Multiple success paths are also a desirable feature of any mitigation system introduced to the plant.

The second and third functions--reactor coolant system (RCS) inventory and pressure control--serve to maintain the coolant over the reactor core. A number of stand-by high-pressure pump systems drawing from several sources of water are available to introduce coolant as required to maintain sufficient inventory and pressure. Multiple relief valves can also act to control excess inventory and/or pressure.

The fourth and fifth core-melt prevention functions serve to remove decay heat from the core and convey it to a point where it can be transferred to the ultimate heat sink. Both functions are required if the core is not to overheat and

TABLE A-2. CLASSES OF SAFETY FUNCTIONS

Class	Functions
1. Anti-core melt	Radioactivity control RCS* inventory control RCS pressure control Core heat removal RCS heat removal
2. Containment integrity	Isolation Pressure and temperature control Combustible gas control
3. Indirect radioactive release	Limit the indirect release of radioactivity
4. Maintenance of vital auxiliaries	Maintenance of vital auxiliaries

*RCS = Reactor Coolant System

Source: W. R. Corcoran et al, "Nuclear Power Plant Safety Functions," Nuclear Safety, 22, 2, March-April 1981.

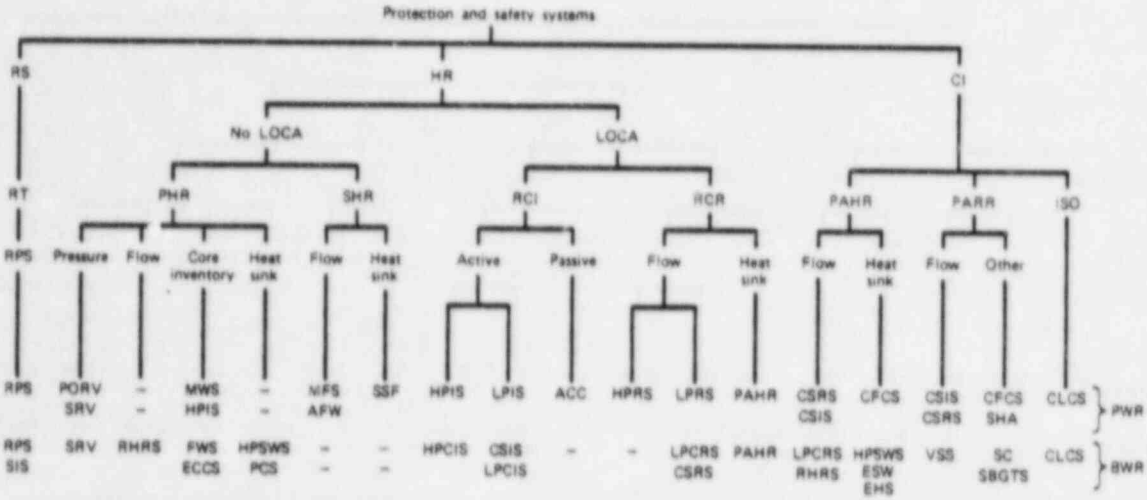
damage itself. One example of RCS heat removal is the so-called 'feed and bleed' concept. This approach to heat removal permits energy to be extracted in the form of steam intentionally released from relief valves. The ultimate heat sink for the decay heat may be a nearby river, lake, cooling pond, or ocean.

A severe accident mitigation system by definition does not have as its primary function the addition or enhancement of core-melt prevention functions, rather it acts after a core melt. Indirectly, the mitigation features may act to reduce the possibility of certain accidents leading to core-melt sequences.

The second safety class pertains to containment integrity and consists of three safety functions. Containment isolation is intended to ensure that all normal containment penetrations are closed in the event of an accident. Activation of the multiple block valves is normally triggered by high containment pressure. Block valves on those lines used for emergency functions are often under manual control only. The presence of missile shields is another element introduced as a protective measure in maintaining containment integrity. The containment pressure and temperature control functions act in concert to maintain these parameters within the design values for the containment. Typically, values range from 15 to 65 psig for the allowable internal pressure and from 200° to 350°F for the allowable temperature. Fan coolers, water sprays, ice reservoirs, mixing and deliberate venting are examples of pressure and temperature control systems found in existing nuclear plants.

Containment overpressurization caused by the deflagration or detonation of hydrogen must be controlled also. Hydrogen can be created by the oxidation of reactor internals (e.g., cladding), oxidation of containment parts, or radiolytic decomposition of water. Combustion control can be accomplished in a number of ways including mixing (for dilution), inerting of the containment with nitrogen, or recombining the hydrogen in a gradual manner with oxygen. Other techniques can be used also. Because of the amount of energy that may be involved and the rate at which it can be released, the hydrogen impact on mitigation systems can be considerable.

Figure A-1 provides a more complete breakdown of the Engineered Safety Features in the prevention of melt and containment integrity classes for both PWRs and BWRs.



General safety functions

CI	Containment Integrity
HR	Heat Removal
RS	Reactor Subcriticality

Condition safety functions

ISO	Containment Isolation
PAHR	Post-Accident Heat Removal
PARR	Post-Accident Radioactivity Removal
PHR	Primary Heat Removal
RCI	Reactor Coolant Injection
RCR	Reactor Coolant Recirculation
RT	Reactor Trip
SHR	Secondary Heat Removal

Typical engineered safety features (ESFs)

PWR

ACC	Accumulators
CFCS	Containment Fan Cooling System
CLCS	Containment Leakage Control System
CSIS	Containment Spray Injection System
CSRS	Containment Spray Recirculation System
ESFCS	Engineered Safety Features, Containment Systems
HPIS	High Pressure Injection System
HPRS	High Pressure Recirculation System
LPIS	Low Pressure Recirculation System
MFS	Main Feedwater System
MWS	Makeup Water System
PORV	Power-Operated Relief Valve
RPS	Reactor Protection System
SHA	Sodium Hydroxide Addition
SRV	Secondary Relief Valves
SSRS	Secondary Steam Relief System

BWR

BIS	Boron Injection System
CLCS	Containment Leakage Control System
CSIS	Core Spray Injection System
CSRS	Core Spray Recirculation System
ECCS	Emergency Core Cooling System
EHS	Emergency Heat Sink
ESFCS	Engineered Safety Features Containment Systems
ESW	Emergency Service Water
FWS	Feedwater System
HPCIS	High Pressure Coolant Injection System
HPSWS	High Pressure Service Water System
LPCIS	Low Pressure Coolant Injection System
LPCRS	Low Pressure Coolant Recirculation System
PCS	Power Conversion System
RCICS	Reactor Core Isolation Cooling System
RHRS	Residual Heat Removal System
RPS	Reactor Protection System
RWCS	Reactor Water Cleanup System
SBGTS	Standby Gas Treatment System
SC	Secondary Confinement
SRV	Secondary Relief Valves
VSS	Vapor Suppression System

Source: L.S. Tong, "Reactor Design and Safety", Nuclear Engineering and Design, 73 (1982), pp. 3-11.

Figure A-1. Reactor core protection and core accident mitigation systems.

The third safety class consists of the single function of controlling indirect radioactive releases. These may occur outside the containment in waste storage facilities or the spent fuel pool for example. Severe accident mitigation systems do not address accidents pertaining to systems located outside the containment.

The fourth and last safety function requires the maintenance of vital auxiliaries used to support other Engineered Safety Features. Examples include the ultimate heat sink, electrical power, control room habitability, instrument air, and safety component cooling water. The extent to which proposed mitigation systems will depend on these auxilaries depends on the specific design of the mitigation system. Ideally, the fewer number of these necessary the more confidence one could place in mitigating severe accidents.

The accident at TMI in March 1979 resulted in formal additions to the criteria which must be satisfied in order to receive a plant construction or manufacturing permit (10 CFR 50.34f, 1982). The emphasis in the new requirements is the enhancement of core-melt prevention and containment integrity functions with the objective of making their operation both more effective and dependable. This is to be achieved through a multiplicity of actions focusing on the detailed characteristics of PWR and BWR Engineered Safety Features. Among other things, it is now necessary that the containment integrity function to be able to accommodate a 100 percent fuel cladding metal/water reaction. Several of the changes called for are important to severe accident mitigation systems. New instrumentation requirements include equipment adequate for monitoring plant conditions following an accident that includes core damage. Also, dedicated penetrations are to be a part of the containment system which may be used at some future date for a containment overpressure protection system. However, the new requirements only apply to those plants whose application was pending February 16, 1982.

APPENDIX B. DESIGN BASIS ACCIDENTS AND PLANT DESIGN CRITERIA

Before receiving a construction permit for a commercial nuclear power plant, the applicant must provide a description of the principal design criteria and a thorough description of the manner in which they can be satisfied. The design criteria establish the design, fabrication, con-XQwchQKro& QnXQKov& eoa ynwYrwieohn wnzcKwninoQX Yrw KQnis important to the overall safety of the plant. These items are provided to ensure that the plant will operate without undue risk to the public.

The Design Basis Accident (DBA) set generally represents the most severe combination of assumed upset conditions for which the applicant need show that the design criteria are satisfied. The DBA set is an outgrowth of operating experience acquired in over 700 reactor-years of operation in the United States. The set varies from plant to plant and depends on site conditions as well as the plant design itself. Some of them are called out specifically in federal regulations while others can only be determined through a joint exercise between the applicant and the NRC staff. Historically, the DBA set has gradually grown in number and in the extent to which it potentially challenges the design criteria. For example, as a result of the TMI accident the amount of hydrogen (produced by metal/water reactions) that the plant must be able to accommodate within the constraints of the design crit[illegible]s increased. A clear understanding of the range [illegible] itions encompassed by the DBA set is helpful in guiding the consideration of mitigation features. The addition of mi[illegible]tion features must extend the plant's ability to accommod[illegible] accidents not otherwise manageable.

The minimum design criteria acceptable in the water-cooled nuclear plant are listed in Appendix A of 10 CFR 50. Currently 53 criteria are identified. These are broken down into the categories of (1) overall criteria, (2) protection by multiple fission-product barriers, (3) protection and reactivity control systems, (4) fluid systems, (5) reactor containment, and (6) fuel and radioactivity control. The criteria are phrased such that they imply or even name the main elements of the DBA set against which they serve as a standard. A listing of some of the more important general design criteria taken from the minimum set is shown in Table B-1.

TABLE B-1. SELECTED GENERAL DESIGN CRITERIA FOR WATER-COOLED REACTORS

I. Overall Requirements
- Quality Standards
- Protection against natural phenomena
- Fire protection

II. Protection of Multiple Fission Product Barriers
- Reactor design
- Reactor inherent protection
- Coolant pressure boundary
- Containment Design
- Electric power systems

III. Protection and Reactivity Control Systems
- Protection System functions
- Failure modes

IV. Fluid Systems
- Quality of reactor coolant pressure boundary
- Reactor coolant makeup
- Residual heat removal
- Emergency core cooling
- Containment heat removal
- Containment atmospheric cleanup

V. Reactor Containment
- Containment design basis
- Piping systems penetrating containment
- Primary containment isolation

VI. Fuel and Radioactivity Control
- Control of releases of radioactive materials to the environment

Source: 10 CFR 50, Appendix A

In a strict sense the DBA set should include all incidents in accident Classes 1 through 8 (see Table A-1 for a definition of all nine accident classes). However, it is normally the case that if the plant can meet the design criteria for those accidents in Class 8, it can be shown to properly accommodate less serious events as well. Before discussing the normal elements of the DBA set, it is worthwhile to note that a core degradation event progressing beyond the reaction of only a small fraction of the cladding with water has not been included in the set. (This is not wholly true since those plants whose application was pending on February 16, 1982, must be able to meet the design criteria even when 100 percent of the cladding reacts with steam and/or water.) The belief has been that if the design criteria can be satisfied for the DBAs to be discussed below, then the likelihood of any accident progressing to severe core damage (and [illegible]y becoming a Class 9 accident) is extremely remot[illegible] [illegible]d unworthy of consideration in the selection of plant features. Consequently, severe accident mitigation has not been required in order to fully satisfy the design criteria.

This situation may be changing at the present time. In April 1983 the NRC announced its proposed policy statement regarding changes in the rules, policies, and regulatory practices that constitute its approach to severe accident rulemaking (see 48 FR 16014). One part of the policy states that an objective of ongoing NRC research is to provide "a technical basis for regulatory decisions to add or modify principal design features and operating guides and procedures of existing plants with respect to their ability to prevent and mitigate severe accidents." In the meantime, other than an additional hydrogen control measure as called out in 46 FR 58484 (December 2,1981), the NRC believes that "individual licensing proceedings are not appropriate forums for a broad examination of the Commission's regulatory requirements relating to control and mitigation of accidents more severe than the design basis." New design criteria and DBA elements may be forthcoming that will incorporate measures to reduce severe accident consequences.

Each of the DBA involves the failure of one or more important plant components or systems. The detailed analysis of the plant's response to these conditions must show that they do not cause conditions to exceed the limits spelled out in the design criteria discussed above, plus whatever additional criteria may be unique to the plant. Therefore, the accidents form the basis for assessing the suitability and acceptability of particular plant and plant site designs vis-a-vis the criteria. For the majority of LWRs the DBAs

are classified on the basis of the relevant initiating mechanism. These mechanisms are:

1. Control rod withdrawal and other transient events.
2. Spent fuel handling.
3. Steam line break.
4. Loss of cooling.
5. External events (e.g., tornado, flood).

B.1 CONTROL ROD WITHDRAWAL AND OTHER TRANSIENT EVENT

The withdrawal of a control rod results in an insertion of positive reactivity. This may occur with the core in a subcritical condition or when it is operating at power (in PWR plants the withdrawal may be the result of a rod housing failure). The power excursion that results from rod withdrawal can lead to fuel failure because the coolant is unable to remove heat sufficiently rapidly from the core. A trip of the reactor system would normally be signaled by the overpower condition itself or by the high coolant temperature or high pressure. A failure to scram would place the reactor in a condition beyond the design basis. In this latter case, the consequences of the accident may be moderated by a severe accident mitigation system. In the absence of a corrective action, the containment could be stressed quickly to its ultimate strength by steam-induced pressure forces. Overpressure failure is possible before the undercooled and subsequently degraded core can melt through the reactor vessel.

A large number of transient events other than control rod withdrawal may occur during the life of the plant and could upset normal plant operations. These events include station blackout, loss of main feedwater, turbine trip, loss of instrument and control air, stuck-open relief valve, spurious closure of the main steam valve, and overpressurization. In many of these instances, action is called for to terminate the thermal output of the reactor. Failures in the sequence of events that follow the initiating transient event raise the possibility of progressively more serious conditions. The choice of the transient events used in the selection of DBAs depends heavily on the details of the plant design itself.

B.2 SPENT FUEL HANDLING

Handling of fuel assemblies takes place with the head of the reactor vessel removed. Radioactivity may be accidently released under this open condition because of mechanical damage caused by the improper handling of the fuel, criticality, or failure to provide adequate cooling to the

assemblies. The plant design is normally arranged to minimize the potential for handling damage, to prevent criticality, and to ensure proper cooling of the fuel at all times. Severe accident mitigation systems could constrain the consequences of those spent-fuel-handling accidents that do take place.

B.3 STEAM LINE BREAK

A steam line break in a PWR creates a rapid cooling of the primary circuit because of the drop in secondary system temperature as its pressure is quickly reduced. This lower temperature creates a "cold-water" reactivity insertion in reactors with a negative power feedback coefficient. The power transient inherent in this event will require actions similar to those taken for control rod withdrawal. The ultimate conditions a mitigation system may need to deal with in the event of a steam line break will be similar to those possible following control rod withdrawal.

B.4 LOSS OF COOLING

A complete loss of cooling is one of the most severe plant accidents. The DBA creating this condition is often assumed to be the double-ended guillotine break of the largest cooling line. Smaller losses of coolant (currently believed to be more likely than a large break) and loss of flow are also significant loss of cooling accidents. Additionally, a loss of heat sink is possible in the event of steam generator failure or loss of feedwater flow to the heat exchanger. The Engineered Safety Features designed to restore cooling quickly are normally activated by high temperature or pressure in the coolant circuit. The various sources of energy that need to be accommodated in order to prevent conditions from exceeding the design criteria include:

1. Sensible heat of the reactor internals.
2. Sensible and latent heat of the coolant.
3. Decay heat.
4. Metal water reactions.

Blowdown forces resulting from flow out the break may cause pipe whip or equipment damage from fluid jet or missile impingement in the event of a major pipe rupture. The ECCS is designed to accommodate those energies inherent in the accident while maintaining the fuel cladding intact. A severe accident mitigation system would be expected to take over the primary safety functions in the event of a coolant loss and failure of the ECCS.

B.5 EXTERNAL EVENTS

The external events selected to be part of the DBA set reflect natural conditions at the plant site. They are a distillation of the geological and hydrological conditions of the site and the historical meteorology of the general area. The parameters that are often selected to characterize these conditions are expressed in terms of maximum values for wind, water level, ground shaking, snow depth, temperature, and humidity. The plant design usually is based on the simultaneous occurrence of extremes in the external events and internal accidents. One combination of this type, for example, is the loss of coolant due to a pipe rupture with a simultaneous loss of off-site power because of an earthquake. Guidance in the selection of a design basis for conditions such as earthquake, flood, and soil instability are provided in 10 CFR 100, entitled Reactor Site Criteria.

It is possible that a decision will be made to include a core degradation accident of some specified extent in the DBA set. This action would acknowledge a heightened awareness of the possibility that the conventional plant safety systems may not adequately arrest the progression of accidents such as loss of coolant. If such action is taken, a number of the existing design criteria would be applicable to the mitigation systems proposed. A listing of these criteria is found in Table B-2. This set of criteria is probably not complete though and may need to be expanded as a part of the severe accident rulemaking process.

TABLE B-2. EXISTING PLANT DESIGN CRITERIA RELEVANT TO MITIGATION SYSTEMS (MS)

Number	Title	Comments
01	Quality Standards and Records	---
02	Design basis for protection against natural phenomena	MS may need to withstand more severe phenomena than normal DBA
03	Fire protection	---
04	Environmental and missile design basis	Steam and/or hydrogen explosions may create missiles
05	Sharing of structures, systems and components	MS should be independent of other protection systems
13	Instrumentation and control	Operator should be able to monitor and manually control (if necessary) the MS
16	Containment design	The MS must not compromise containment integrity
19	Control room	See Number 13
36	Quality of reactor coolant pressure boundary	The MS must not compromise the pressure boundary
38	Containment heat removal	Heat removal must be possible under severe accident conditions
39	Inspection of containment heat removal systems	---
40	Testing of containment heat removal system	---

TABLE B-2. CONTINUED

Number	Title	Comments
41	Containment atmosphere cleanup	Recovery from a severe accident requires allowance for eventual cleanup
44	Cooling water	Provisions must exist in MS to transfer energy to the ultimate sink. Cooling water is a likely choice.
45	Inspection of cooling water system	---
46	Testing of cooling Water system	---
50	Containment design basis	MS may incorporate methods of strengthening the containment
51	Fracture prevention of containment pressure boundary	---
52	Capability of containment leak rate testing	---
53	Provisions for containment testing and inspection	---
54	Piping systems penetrating containment	Isolation capability required
56	Primary containment isolation	Isolation capability required
57	Closed system isolation valves	Isolation capability required
60	Control of releases of radioactive materials to the environment	Focuses on the primary objective of the MS

APPENDIX C. REACTOR CONTAINMENT FAILURE MODES

This appendix describes potential containment failure modes pertinent to severe accident assessment and the possible causes of such failures. Containments of interest are those presently in use at U.S. LWR plants as well as those under construction or being considered for construction (see Appendix D and E). These containments include the large dry type, the subatmospheric type, and the pressure-suppression type (i.e., ice condenser, Mark I, II and III). Forms of construction for containments include free-standing steel shells and steel-membrane-lined concrete. An understanding of the various challenges to containment integrity serves as a guide in the selection of severe accident mitigation features.

For the purpose of this discussion, containment failure occurs when the leak rate significantly exceeds the design leak rate. A design leak rate of about 0.1 to 0.5 percent of the internal volume per day at the design pressure is typical. The primary containment leakage barrier is normally formed by heavily reinforced concrete, steel plates, or some combination of the two materials. Provisions are included in the plant design so that the containment leakage rate can be verified periodically (as called for in Appendix J of 10 CFR 50). Hundreds of piping, electrical, ducting and access penetrations pass through the containment walls and are potential isolation failure paths. The elements of the containment system that may be relevant to the ability of it to withstand severe accident challenges are listed in Table C-1. Typical containment features appear in Tables C-2 and C-3, also in Appendix D and E.

Several potential causes for loss of the containment function have been identified. These causes can be combined into a few generic classes that are listed in Table C-4. Each will be discussed below.

C.1 SEVERE ACCIDENT INITIATING EVENT

An incident capable of causing or serving as a precursor to an accident beyond the design basis may be responsible for the loss of containment capability as well. For example, a large earthquake could rupture the coolant system boundary and destroy the multiple emergency core cooling functions. At the same time, ground shaking could cause gross structural failure or cracking of the containment. Also, shaking could cause penetrations to tear out of the containment wall. Other examples of initiating events capable of damaging the containment are sabotage, tornado-generated missiles, and

TABLE C-1. CONTAINMENT FEATURES PERTINENT TO CONTAINMENT FAILURE MODES

- Free volume
- Design pressure and ultimate pressure
- Type of concrete aggregate
- Presence of water
- Concrete thickness in the basemat
- Missile shields
- Seismic rating
- Failure temperature of penetration seals and equipment
- Design leak rate
- Manual and automatic isolation features
- Atmospheric cooling provisions
- Inherent ignition features
- Presence of secondary containment or shield building
- Ability to withstand external pressure
- Interfacing LOCA paths

TABLE C-2. TYPICAL CONTAINMENT VOLUMES

Type	Example Plant	Free Volume ($10^6 ft^3$)
Large Dry		
Prestressed concrete	TMI-2	2.0
Free-standing steel	St. Lucie	2.5
Subatmospheric, reinforced concrete	Surry	1.8
Spherical steel shell	Perkins	3.3
Pressure Suppression		
Ice condenser	Sequoyah	1.2
Mark I	Peach Bottom	0.28
Mark II	Zimmer	0.39
Mark III	Grand Gulf	1.7

TABLE C-3. CONTAINMENT DESIGN PRESSURES (PSIG)

June, 1980 data base			Containment structures						
			Concrete			Steel			
			Prestressed vertical cylinder and dome with flat base	Deformed bar hemispherical dome, vertical cylinder and flat base	Other concrete	Light bulb form containment	Steel sphere	Hemispherical dome, cylindrical body and ellipsoidal base	Hybrid steel dome and body, reinforced concrete base
U.S. reactors	PWR	Atmospheric w/o suppression	47-60	42-55	42-60		25-48	34-44	
		Subatmospheric		45					
		Ice condenser		12					11-15
	BWR	Mark I			56	40-56			
		Mark II			45-56				
		Mark III		15					15
		Pre-mark			52		27-30		

Source: T. E. Blejwas et al., 1982

TABLE C-4. GENERIC CAUSES OF CONTAINMENT FAILURE

Cause	Designation in Reactor Safety Study(WASH-1400)
Severe accident-initiating event (e.g., earthquake, tornado, etc.)	---
Internal missiles	α
High temperature	---
Basemat melt through	ε
Slow or rapid overpressurization due to:	
Steam spike or steam explosion	α
Steam	δ
Hydrogen burning or detonation	γ
Non-condensible gas	δ
Failure to isolate	β
Interfacing LOCA/Bypass	ν

tsunamis. Without a containment, a severe accident may cause the release of radioactive materials to the atmosphere much more quickly and consequently, in a larger dose, than would otherwise be the case. The importance of radioactivity scrubbing functions within the plant increases dramatically in the event of containment failure.

C.2 INTERNAL MISSILES

The possible rapid conversion of thermal to mechanical energy within or outside the reactor vessel raises a concern about missiles generated by steam forces. Missiles represent a threat to the containment and could cause failure by direct, high-velocity impingement or by indirect damage to penetrations and/or isolation valves. Candidate missiles include the vessel head, control rods, vessel nozzles, and other materials in and around the vessel.

Very rapid transfer of energy between the core debris and the coolant is referred to as a steam explosion. Explosions of this type have occurred in the steel and aluminum industries when molten metals were accidently dropped into pools of water. Small-scale steam explosions have also be produced in laboratory-scale experiments using reactor simulant materials at low and high temperatures. The process requires the presence of a suitable premixture in which hot material is finely divided and immersed in the coolant. The fuel coolant mixture must be sufficiently coarse to allow--at least initially--thermal isolation between the materials and fine enough to support propagation of the energy exchange process. The theoretical conversion ratio is roughly 30 percent (Corradini, 1982). It is possible for extremely rapid heat transfer to cause an explosive generation of steam. The explosion can damage equipment and may propel broken items forcefully against the containment. The quantity of thermal energy contained within the molten fuel which may be partially converted into mechanical energy can be as large as 200,000,000 Btu if the core is fully involved in a meltdown accident. Figure C-1 illustrates how a steam explosion could cause the reactor head to become a containment-threatening missile.

There are two situations in which a steam explosion may occur: one in the vessel and the other outside. In-vessel conditions may be suitable for an energetic reaction if appreciable molten core materials drop into a pool of coolant held in the lower vessel head. Similar ex-vessel conditions are present if molten materials drop out of the vessel and into a pool of water in the containment below the vessel.

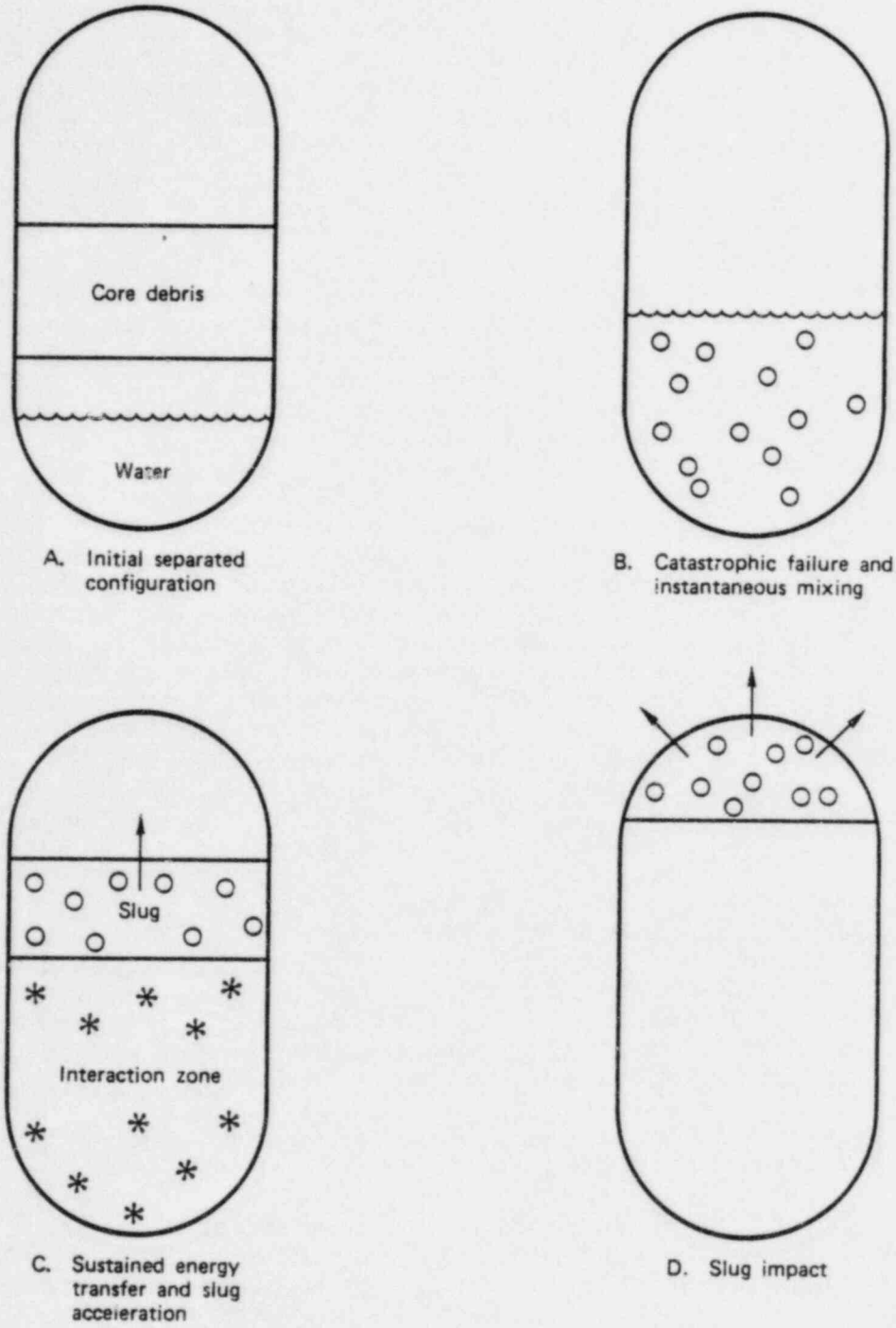


Figure C-1 Missile generation by steam-explosion induced slug impact.

Steam explosions have also been observed to occur when coolant is introduced on top of molten materials similar to corium. In this case, as with the situation in which molten materials drop into the coolant, the important mechanisms driving the reaction are not fully understood. Some severe reactor accidents may result in molten core materials leaving the vessel with water then falling on top of the mass. This situation may lead to a steam explosion and/or missile generation.

The susceptibility of the containment to missiles is partially dependent upon the presence of missile barriers within it and other equipment that may deflect and/or absorb missiles and their associated kinetic energy. If safety items, such as containment sprays, are damaged, their usefulness in mitigating accident consequences would be impaired.

C.3 HIGH TEMPERATURE

The controlling sensitivity to temperature damage in the containment barrier often concerns the seals around various structural penetrations (i.e., pipes, ducts, hatches and electrical cables). The elastomeric materials sometimes used to form a pressure-tight seal lose their retention qualities at temperatures as low as 200° to 300°F. Heating of these seals during a severe accident could be by radiation from ex-vessel core debris, convection in a hot containment atmosphere, conduction from adjacent materials, or by condensation of steam at elevated containment pressures. During normal plant operations and DBAs, the temperature of the containment atmosphere is controlled through the use of fan coolers (with cooling water provided from outside the containment), ice condensers, suppression pools, and/or water sprays into the containment atmosphere. These Engineered Safety Features may not be available in an accident characterized by core degradation, however.

C.4 BASEMENT PENETRATION

Molten core debris and other reactor internals may be released from the vessel as one consequence of a core degradation accident. The quantity of this material--known as corium--may be as large as 100 tons of fuel and 100 tons of cladding and steel if the core is fully involved in a meltdown. Corium could drop or dribble through a breach in the thermally weakened and partially melted lower vessel head or may be forcefully ejected if the vessel is still pressurized at the time of breakthrough. Once outside the reactor vessel, the corium would begin to thermally attack whatever surface it came to rest on and strongly radiate thermal

energy to other surfaces if not immersed in coolant. The core materials and steel may both fragment into fine irregular shaped particles if plunged into a water pool in the reactor cavity or pedestal area. Thermal radiation may weaken and cause failure of steel components in the containment.

The decay heat rate of the corium may be as large as several percent of the reactor's full power rating. Rates as high as 25 MW are possible. If this energy is not extracted by a coolant or some other means, it is capable of seriously eroding the concrete structure of the containment. Figure C-2 shows conceptually how a hot mass of corium may move downward and radially outward through a supporting concrete layer. The thermal energy can be sufficiently large to gasify parts of the concrete and liquify others. The gaseous products generated depend on the nature of the concrete aggregate. Siliceous, calcareous and basaltic types are common. Strong heating of limestone-type aggregate (calcareous) creates copious amounts of CO_2 ($CaCO_3$ $CaO + CO_2$). Free and chemically bound water is released from all concrete types when heated above 200°F. The reaction of gases driven from concrete with the hot corium constituents can create other noncondensible and potentially hazardous gases. CO, H_2 and CH_2 are possible reaction products (e.g., $3H_2O + 2Fe \longrightarrow Fe_2O_3 + 3H_2$). These chemical reactions are all facilitated and highly efficient if the corium is in a liquid state and the gases originating in the concrete bubble through it. Even at its freezing temperture and lower temperatures, the corium is still able to melt concrete. The thickness of the concrete basemat in the reactor containment is normally about 5 to 10 ft and its melting temperature is about 1100°C. Corium with a temperature as high as 3000° to 4000°C is capable of melting downward through the concrete basemat in approximately 1 to 3 days if no cooling mechanism is otherwise available. Even an overlying pool of water could have its cooling capabilities greatly hindered if a solid crust were to form on the upper surface of the corium.

Two analytic tools used to predict the rate of concrete penetration by corium are the INTER and CORCON codes. Extensive study of penetration rates has also been carried out by contractors for the West German nuclear program. Failure of the containment would allow core materials and containment atmospheric products to enter the soil under the plant. The materials could move downward into underlying water tables or--if in a gaseous state--move through porous soil and rock upward to the surface and into the atmosphere. Specific site conditions would establish the relative amounts of the various types of migrations possible in the

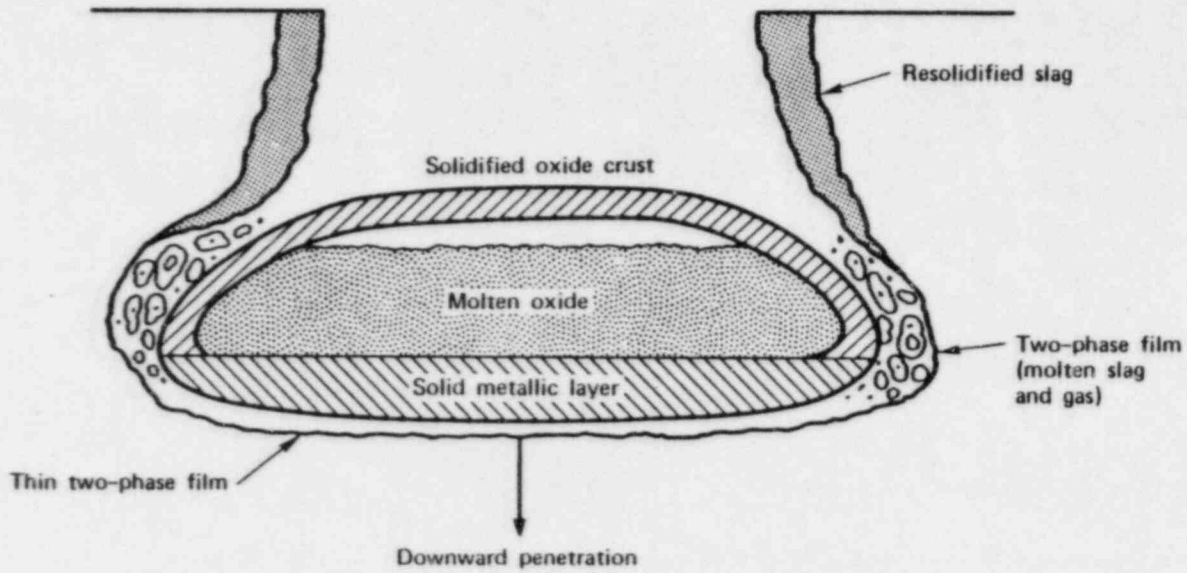


Figure C-2. Conceptual picture of core in concrete after meltdown.

earth. The soil would act to remove some fission products no matter which direction the materials may move. The gradual dispersion of the corium would eventually bring it to rest beneath the plant.

C.5 OVERPRESSURIZATION

Several mechanisms may act to create containment pressures in excess of the design pressure and the containment ultimate strength in the event of a core degradation accident. The pressure may rise quickly or not depending on whether steam is produced by a steam explosion or steam spike or through more gradual boiling of the water coolant. Rapid overpressurization is possible also if a detonation or deflagration (i.e., very rapid burning) of hydrogen takes place in the containment. Hydrogen can be made available in potentially large quantities from metal/water reactions. Even without detonation or deflagration, the evolution of hydrogen or other noncondensible gases driven from concrete structures by strong heating could cause excessive containment pressure.

The ability to accommodate containment pressurization mechanisms is a function of the containment ultimate strength and free volume. These two parameters vary considerably among the various containment types. For example, a large dry containment may have a design pressure of 45 psig and a free volume of 2,000,000 ft^3 while a pressure-suppression-type containment might be characterized by a 60-psig design pressure and 300,000 ft^3. The product of the pressure and volume is a rough indicator of the containment's ability to accommodate gas production. The capacities vary by a factor of five in the two examples just given.

C.5.1 Steam Explosion/Steam Spike

Section C.2 describes how a steam explosion might take place involving molten core debris and coolant. The shock wave created by any such explosion and/or the steam produced in the explosion represents a possible threat to the containment structure, particularly in the case of an ex-vessel explosion. Failure may take the form of gross cracking, blow-out of penetrations, or structural collapse. It is expected that failure would occur at a pressure two to three times the design pressure.

C.5.2 Steam

Gradual overpressurization of the containment by steam is also a possibility in an accident that surpasses the design

basis. The most serious challenge of this type is presented by an ATWS event. The rate of power production in the core can be as large as 20 percent of full power during an ATWS with no load available. Consequently, large quantities of steam would be released through primary system relief valves to the containment. Without a containment heat removal system of large capacity, the design pressure would be quickly exceeded. Consider this example case. Suppose a reactor with a nominal thermal output of 3000 MWt were generating exclusively steam at a 20 percent power level. Roughly $2x10^9$ Btu/hr would be converted to steam. At 1000 psig the latent heat of evaporation of water is 650 Btu/lb. Therefore, the steam production rate could be as high as 50,000 lb/min in the absence of any heat removal processes.

Water vapor is also available from heated concrete in the event hot core debris were to breach the reactor vessel and drop into the containment. The quantity released depends on the extent to which the concrete is heated. This is difficult to predict because of uncertainties in geometry and phenomenology. However, a large amount of free and chemically bound water is held in the concrete and is potentially available for containment pressurization under the proper conditions (roughly 5 percent of the total concrete mass is made up of water).

C.5.3 Hydrogen Burning/Detonation

Hydrogen can be introduced into the containment from several sources. The first of these occur from high-temperature, metal/water reactions within the vessel. Both fuel cladding and steel internal react exothermically with high-temperature steam and/or water to produce hydrogen. (In the TMI-2 accident, a cladding reaction produced about 500 kg of hydrogen, which was subsequently released to the containment where it ignited.) Complete oxidation of the cladding is capable--if it occurs--of producing several thousand kilograms of hydrogen. Hydrogen release to the containment can be through safety/relief valves or breaks in the vessel or associated piping.

Ex-vessel core-concrete interactions represent another poten-tial--and possibly major--source of hydrogen. Water is driven from the concrete when it is heated. This water may result in further oxidation of corium materials if the water interacts with very hot cladding or steel. Thousands of kilograms of hydrogen could be rapidly generated by oxidation of large quantities of steel if molten corium were to rest on the concrete basemat for a protracted period.

Minor amounts of hydrogen are also available from the radiolytic decomposition of water and oxidation of some paints. The plant may be equipped with a small capacity hydrogen/oxygen recombiner to routinely dispose of radiolytic hydrogen during normal operations. Recombiners are likely to be unavailable and/or of insufficient capacity to be useful in a core degradation accident.

Burning or detonation of hydrogen requires the proper fuel/oxidizer ratio plus an ignition source. Deliberate ignition sources (glow plugs) may be present, or electrical equipment may provide ignition. The fuel/oxidizer ratio may be highly uneven throughout the containment and may be too rich or too lean to support combustion. Steam, for example, can serve to maintain a noncombustible atmosphere. Nonuniformities in hydrogen concentration are aggravated if the containment is compartmentalized.

The deflagration of a stoichiometric hydrogen-air mixture initially at 1 atm results in a pressure of approximately 8 atm. This occurrence has the potential for threatening the integrity of all containment types if it were to involve appreciable amounts of hydrogen. Hydrogen threats have received considerable attention since the TMI-2 accident. The use of glow plugs and nitrogen inerting of the small Mark I and II containments are two consequences of post-TMI actions. Glow plugs serve to cause the continuous burning of hydrogen as it is released to the containment rather than allowing it to accumulate and detonate. The consequences of this gradual burning are more manageable than would be the case if a large amount of hydrogen were to explode.

C.5.4 Noncondensible Gases (H_2, CO_2, CO, CH_4)

Several types of noncondensible gases may be created and released to the containment during a severe accident characterized by core degradation. Hydrogen is one of these gases as discussed above. Others may appear when concrete portions of the containment are heated to temperatures in excess of 200°F. Free and chemically bound water can be driven from the concrete at elevated temperature. Inasmuch as roughly 5 percent of the concrete mass is water, large quantities are potentially available for release. If the water vapor is involved in an oxidation reaction with the hot corium responsible for concrete heating, then hydrogen again is produced.

Methane and other alkanes are also noncondensible gases that may appear in the containment. They are produced by the reaction of carbon dioxide and hydrogen at temperatures on the order of 300°C (Swanson, 1983) if metal catalysts are

presently (which is likely to be the case). Carbon dioxide is a concrete decomposition product. Heating of limestone aggregate creates carbon dioxide. The carbon dioxide can be reduced to another noncondensible--carbon monoxide--if it is involved in a metal oxidation reaction. The corium provides an opportunity for this reaction when its temperature is in excess of roughly 900°C. Other aggregate types do not produce nearly the same quantity of carbon dioxide as does limestone.

Several of the oxidation reactions involving corium are exothermic in nature. Reaction energies combine with fission-product decay heat as a driving force in the accident scenario.

C.6 FAILURE TO ISOLATE/INTERFACING LOCA/BYPASS

The containment integrity function is circumvented if containment penetrations are not properly isolated in the event of a severe accident. The integrity function is also defeated if a piping failure occurs which allows radioactive material releases outside the containment boundary. Redundant isolation valves or dampers are included as part of the piping and ventilation penetrations in order to reduce the likelihood of such isolation failures. Furthermore, these valves and dampers are normally of a fail-safe type. The large number of penetrations and the fact that some must stay open to accomplish safety functions increases the danger of not properly isolating the containment when necessary. A good many incidents have been reported (D. Moeller, 1983) wherein the automatic closure feature of isolation valves was lost or the valves were locked open. The containment boundary is also breached when failures occur in the steam generator tubes of a PWR plant.

The release of radioactive materials through or outside the containment in the event of a core degradation accident complicates the task of mitigating the accident consequences In instances of this type, it may be prudent and of greater value to concentrate on prevention of such occurrences.

APPENDIX D

REACTOR CONTAINMENT SYSTEMS

The following table lists all the U. S. nuclear power plants, grouped according to type of containment. For each plant the name and location are given, also the plant size, the heat sink or nearby body of water, date of licensing, and comments about the containment structure.

REACTOR CONTAINMENT SYSTEMS

1. DRY CONTAINMENT (PWR[+])

Plant name	Location	MW_e	Nearby water	Date of operating license	Type of construction	Concrete secondary containment	Steel secondary containment	Comments
Yankee Rowe	Rowe, MA	175	Deerfield River	6-23-61	S			Steel sphere
Big Rock Point	Charlevoix City, MI	75	Lake Michigan	5-1-64	S			BWR, steel sphere
San Onofre 1	Camp Pendleton, CA	430	Pacific Ocean	3-27-67	S	Yes		Steel sphere
Haddam Neck	Haddam Neck, CT	575	Connecticut River	12-27-74	RC			Concrete cylinder with steel liner
Ginna 1	Ontario, NY	490	Lake Ontario	6-19-69	PT			Concrete cylinder with steel liner
Indian Pt. 2	Buchanan, NY	873	Hudson River	10-19-71	RC			Concrete cylinder with steel liner
Turkey Pt. 3	Dade County, FL	693	Biscayne River	7-19-72	PT			Concrete cylinder with steel liner
Turkey Pt. 4	Dade Coutny, FL	693	Biscayne River	4-10-73	PT			Concrete cylinder with steel liner
Palisades	Covent Township, MI	805	Lake Michigan	3-24-71	PT			Concrete cylinder with steel liner
Robinson 2	Hartsville, SC	700	Robinson Reservoir	7-31-70	PT			Concrete cylinder with steel liner
Point Beach 1	Manitowoc County, WI	497	Lake Michigan	10-5-70	PT			Concrete cylinder with steel liner
Oconee 1	Oconee County, SC	887	Keowee Lake	2-6-73	PT			Concrete cylinder with steel liner
Oconee 2	Oconee County, SC	887	Keowee Lake	10-6-73	PT			Concrete cylinder with steel liner

DRY CONTAINMENT (PWR+) - CONTINUED

Plant name	Location	MW_e	Nearby water	Date of operating license	Type of construction	Concrete secondary containment	Steel secondary containment	Comments
Salem 1	Salem, NJ	1090	Delaware River	8-13-76	RC			Concrete cylinder with steel liner
Diablo Canyon 1	Diablo Canyon, CA	1084	Pacific Ocean	--	RC			Concrete cylinder with steel liner
Prairie Island	Red Wing, MN	530	Mississippi River	8-9-73	S	Yes		Steel cylinder
Fort Calhoun 1	Washington City, NE	457	Missouri River	5-24-73	PT			Concrete cylinder with steel liner
Indian Point 3	Buchanan, NY	873	Hudson River	12-12-75	RC			Concrete cylinder with steel liner
Oconee 3	Oconee City, SC	887	Keowee Lake	7-19-74	PT			Concrete cylinder with steel liner
3 Mile Island 1	3 Mile Island, PA	819	Susquehanna River	4-19-74	PT			Concrete cylinder with steel liner
Zion 1	Zion, IL	1040	Lake Michigan	4-6-73	PT			Concrete cylinder with steel liner
Point Beach 2	Manitowoc County, WI	497	Lake Michigan	10-16-71	PT			Concrete cylinder with steel liner
Crystal River 3	Crystal River, FL	802	Gulf of Mexico	12-33-76	PT			Concrete cylinder with steel liner
Zion 2	Zion, IL	1040	Lake Michigan	11-14-73	PT			Concrete cylinder with steel liner
Kewaunee	Carlton, WI	535	Lake Michigan	12-21-73	S	Yes		Steel cylinder
Prairie Island 2	Red Wing, MN	530	Mississippi River	10-29-74	S	Yes		Steel cylinder
Maine Yankee	Lincoln County, ME	790	Back River	9-15-71	RC			Concrete cylinder with steel liner

DRY CONTAINMENT (PWR*) - CONTINUED

Plant name	Location	MW_e	Nearby water	Date of operating license	Type of construction	Concrete secondary containment	Steel secondary containment	Comments
Salem 2	Salem, NJ	1115	Delaware River	4-18-80	RC			Concrete cylinder with steel liner
Rancho Seco	Sacramento County, CA	918	Folsom Canal	8-16-74	PT			Concrete cylinder with steel liner
Arkansas Nuclear 1	Pope City, AR	850	Dardanelle Reservoir	5-21-74	PT			Concrete cylinder with steel liner
Calvert Cliffs 1	Lusby, MD	845	Chesapeake River	7-31-74	PT			Concrete cylinder with steel liner
Calvert Cliffs 2	Lusby, MD	845	Chesapeake River	6-13-76	PT			Concrete cylinder with steel liner
Diablo Canyon 2	Diablo Canyon, CA	1106	Pacific Ocean	--	RC			Concrete cylinder with steel liner
Midland 1	Midland, MI	460	Cooling Lake	--	PT			Concrete cylinder with steel liner
Midland 2	Midland, MI	811	Cooling Lake	--	PT			Concrete cylinder with steel liner
Millstone 2	Waterford, CT	830	Long Island Sound	8-1-75	PT		Yes	Concrete cylinder with steel liner
St. Lucie 1	Hutchinsons Isl, FL	802	Atlantic Ocean	3-1-76	S	Yes		Steel cylinder
Trojan	Columbia, OR	1130	Columbia River	11-21-75	PT			Concrete cylinder with steel liner
Davis Besse 1	Ottawa County, OH	906	Lake Erie	4-22-77	S	Yes		Steel cylinder
Farley 1	Dothan, AL	829	Woodruff Reservoir	6-25-77	PT			Concrete cylinder with steel liner
San Onofre 2	Camp Pendleton, CA	1100	Pacific Ocean	2-16-82	PT			Concrete cylinder with steel liner
San Onofre 3	Camp Pendleton, CA	1100	Pacific Ocean	11-15-82	PT			Concrete cylinder with steel liner

DRY CONTAINMENT (PWR*) - CONTINUED

Plant name	Location	MW_e	Nearby water	Date of operating license	Type of construction	Concrete secondary containment	Steel secondary containment	Comments
Farley 2	Dothan, AL	829	Woodruff Reservoir	10-23-80	PT			Concrete cylinder with steel liner
Arkansas 2	Pope County, AL	912	Dardanelle Reservoir	7-18-78	PT			Concrete cylinder with steel liner
Waterford 3	Taft, LA	1113	Mississippi River	--	S	Yes		Steel cylinder
St. Lucie 2	Hutchinsons Isl, FL	810	Atlantic Ocean	4-6-83	S	Yes		Steel cylinder
Summer 1	Broad River, SC	900	Lake Mont.	8-6-82	PT			Concrete cylinder with steel liner
Harris 1	Bonsal, NC	900	Cape Fear River	--	RC			Concrete cylinder with steel liner
Harris 2	Bonsal, NC	900	Cape Fear River	--	RC			Concrete cylinder with steel liner
LaCrosse (BWR)	Genoa, WI	50	Mississippi River	8-28-73	S			BWR, steel cylinder
Vogtle 1	Waynesboro, GA	1113	Savannah River	--	PT		Yes	Concrete cylinder with steel liner
Vogtle 2	Waynesboro, GA	1113	Savannah River	--	PT		Yes	Concrete cylinder with steel liner
Bellfonte 1	Scottsboro, AL	1213	Tennessee River	--	PT	Yes		Concrete cylinder with steel liner
Seabrook 1	Seabrook, NH	1200	Atlantic Ocean	--	RC	Yes		Concrete cylinder with steel liner
Seabrook 2	Seabrook, NH	1200	Atlantic Ocean	--	RC	Yes		Concrete cylinder with steel liner
Bellfonte 2	Scottsboro, AL	1213	Tennessee River	--	PT	Yes		Concrete cylinder with steel liner

DRY CONTAINMENT (PWR*) - CONTINUED

Plant name	Location	MW_e	Nearby water	Date of operating license	Type of construction*	Concrete secondary containment	Steel secondary containment	Comments
Comanche Peak 1	Glen Rose, TX	1150	Squaw Creek Reservoir	--	RC			Concrete cylinder with steel liner
Comanche Peak 2	Glen Rose, TX	1150	Squaw Creek Reservoir	--	RC			Concrete cylinder with steel liner
Byron 1	Byron, IL	1120	Rock River	--	PT			Concrete cylinder with steel liner
Byron 2	Byron, IL	1120	Rock River	--	PT			Concrete cylinder with steel liner
Braidwood 1	Braidwood, IL	1120	Kankakee River	--	PT			Concrete cylinder with steel liner
Braidwood 2	Braidwood, IL	1120	Kankakee River	--	PT			Concrete cylinder with steel liner
Washington 1	Richland, WA	1218	Columbia River	--	RC			Concrete cylinder with steel liner
Wolf Creek	Burlington, KS	1150	Wolf Creek	--	PT			Concrete cylinder with steel liner
Calloway 1	Fulton, MO	1120	Missouri River	--	PT			Concrete cylinder with steel liner
Cherokee 1	Cherokee County, SC	1280	Broad River	--	S	Yes		Steel sphere
South Texas 1	Bay City, TX	1250	Colorado River	--	PT			Concrete cylinder with steel liner
South Texas 2	Bay City, TX	1250	Colorado River	--	PT			Concrete cylinder with steel liner
Washington 3	Satsop, WA	1242	Chehalis River	--	S	Yes		Steel cylinder

DRY CONTAINMENT (PWR+) - CONTINUED

Plant name	Location	MW_e	Nearby water	Date of operating license	Type of construction*	Concrete secondary containment	Steel secondary containment	Comments
Palo Verde 1	Wintersberg, AZ	1238	Phoenix sewage	--	PT			Concrete cylinder with steel liner
Palo Verde 2	Wintersberg, AZ	1238	Phoenix sewage	--	PT			Concrete cylinder with steel liner
Palo Verde 3	Wintersburg, AZ	1238	Phoenix sewage	--	PT			Concrete cylinder with steel liner
Marble Hill 1	Madison, IN	1130	Ohio River	--	PT			Concrete cylinder with steel liner
Marble Hill 2	Madison, IN	1130	Ohio River	--	PT			Concrete cylinder with steel liner
Yellow Creek 1	Corinth, MS	1285	Pickwick Reservoir	--	S	Yes		Steel sphere
Yellow Creek 2	Corinth, MS	1285	Tennessee River	--	S	Yes		Steel sphere
Total: 76 plants								

NOTES: + Unless noted otherwise
* S = Steel
RC = Reinforced concrete
PT = Post-tensioned concrete

2. ICE CONDENSER (PWR)

Plant name	Location	MW_e	Nearby water	Date of operating license	Type of construction	Concrete secondary containment	Comments
Cook 1	Benton Harbor, MI	1054	Lake Michigan	10-25-74	RC		Concrete cylinder with steel liner
Cook 2	Benton Harbor, MI	1064	Lake Michigan	12-23-77	RC		Concrete cylinder with steel liner
Sequoyah 1	Daisy, TN	1148	Chickamauga Lake	10-23-80	S	Yes	Steel cylinder
Sequoyah 2	Daisy, TN	1148	Chickamauga Lake	6-25-81	S	Yes	Steel cylinder
Watts Bar 1	Spring City, TN	1177	Chickamauga Lake	--	S	Yes	Steel cylinder
Watts Bar 2	Spring City, TN	1177	Chickamauga Lake	--	S	Yes	Steel cylinder
Catawba 1	Lake Wylie, SC	1153	Lake Wylie	--	S	Yes	Steel cylinder
Catawba 2	Lake Wylie, SC	1153	Lake Wylie	--	S	Yes	Steel Cylinder
McGuire 1	Cowans Ford Dam, NC	1180	Lake Norman	1-28-81	S	Yes	Steel Cylinder
McGuire 2	Cowans Ford Dam, NC	1180	Lake Norman	3-3-83	S	Yes	Steel Cylinder
Total: 10 plants							

3. SUBATMOSPHERIC (PWR)

Plant name	Location	MW_e	Nearby water	Date of operating license	Comments
Surry 1	Surry County, VA	822	James River	2-25-76	Concrete cylinder with steel liner
Surry 2	Surry County, VA	822	James River	1-29-73	Concrete cylinder with steel liner
North Anna 1	Louisa County, VA	898	Cooling Lake	11-26-77	Concrete cylinder with steel liner
North Anna 2	Louisa County, VA	907	Cooling Lake	4-11-80	Concrete cylinder with steel liner
Beaver Valley 2	Shippingport, PA	852	Ohio River	--	Concrete cylinder with steel liner
Millstone 3	Waterford, CN	1156	Long Island Sound	--	Concrete cylinder with steel liner
Beaver Valley 1	Shippingport, PA	852	Ohio River	1-30-76	Concrete cylinder with steel liner
Total: 7 plants					

4. MARK I (BWR) - STEEL DRYWELL AND STEEL WETWELL/CONCRETE SECONDARY CONTAINMENT

Plant name	Location	MW_e	Nearby water	Date of operating license	Comments
Oyster Creek 1	Oyster Creek, NJ	650	Barnegat River	4-9-69	
Nine Mile Point 1	Oswego, NY	610	Lake Ontario	12-26-74	
Dresden 2	Grundy County, IL	794	Kankakee River	12-22-69	
Millstone 1	Waterford, CT	660	Long Island Sound	10-7-70	
Dresden 3	Grundy County, IL	794	Kankakee River	1-12-71	
Quad Cities 1	Rock Island, IL	789	Mississippi River	10-1-71	
Browns Ferry 1	Decatur, GA	1065	Tennessee River	6-26-73	
Browns Ferry 2	Decatur, GA	1065	Tennessee River	6-28-74	
Monticello	Monticello, MN	545	Mississippi River	1-9-71	
Quad Cities 2	Rock Island, IL	789	Mississippi River	3-31-72	
Vermont Yankee	Vernon, VT	514	Connecticut River	3-21-72	
Peach Bottom 2	York County, PA	1065	Susquehanna River	8-8-73	
Peach Bottom 3	York County, PA	1065	Susquehanna River	7-2-74	
Pilgrim 1	Plymouth, MA	655	Cape Cod Bay	6-8-72	
Browns Ferry 3	Decatur, GA	1065	Tennessee River	7-2-76	
Brunswick 1	Brunswick County, NC	821	Cape Fear River	9-8-76	Reinforced concrete
Brunswick 2	Brunswick County, NC	821	Atlantic Ocean	12-27-74	Reinforced concrete
Cooper	Nemaha County, NE	778	Missouri River	1-18-74	
Hatch 1	Baxley, GA	786	Atamaha River	8-6-74	

MARK I (BWR) - STEEL DRYWELL AND STEEL WETWELL/CONCRETE SECONDARY CONTAINMENT - CONTINUED

Plant name	Location	MW_e	Nearby water	Date of operating license	Comments
Arnold	Cedar Rapids, IA	538	Cedar Rapids River	2-22-74	
Fitzpatrick	Oswego, NY	821	Lake Ontario	10-17-74	
Fermi 2	Newport, MI	1093	Lake Erie	--	
Hope Creek 1	Salem, NJ	1067	Delaware River	--	
Hatch 2	Baxley, GA	795	Altamaha River	6-13-78	
Total: 24 plants					

5. MARK II (BWR) - CONCRETE DRYWELL AND CONCRETE WETWELL WITH STEEL LINER, CONCRETE SECONDARY CONTAINMENT

Plant name	Location	MW	Nearby water	Date of operating license
Shoreham	Brookhaven, NY	819	Long Island Sound	--
Limerick 1	Pottstown, PA	1065	Schuykell River	--
Limerick 2	Pottstown, PA	1065	Schuykell River	--
Zimmer 1	Moscow, OH	810	Ohio River	--
La Salle 1	Seneca, IL	1078	Illinois River	4-17-82
La Salle 2	Seneca, IL	1078	Illinois River	--
Susquehanna 1	Berwick, PA	1050	Susquehanna River	7-17-82
Susquehanna 2	Berwick, PA	1050	Susquehanna River	--
Nine Mile Point 2	Scriba, NY	1100	Lake Ontario	--
Washington 2	Satsop, WA	1100	Columbia River	--
Total: 10 plants				

6. MARK III (BWR) - CONCRETE DRYWELL AND CONCRETE WETWELL WITH STEEL LINER, CONCRETE SECONDARY CONTAINMENT

Plant name	Location	MW_e	Nearby water	Date of operating license	Comments
Grand Gulf 1	Port Gibson, MS	1150	Mississippi River	6-16-82	
Grand Gulf 2	Port Gibson, MS	1150	Mississippi River	--	
Perry 1	Perry, OH	1205	Lake Erie	--	Steel primary containment
Perry 2	Perry, OH	1205	Lake Erie	--	Steel primary containment
River Bend 1	St. Francisville, LA	934	Mississippi River	--	Steel primary containment
River Bend 2	St. Francisville, LA	934	Mississippi River	--	Steel primary containment
Clinton 1	Clinton, IL	933	Salt Creek Reservoir	--	
Clinton 2	Clinton, IL	933	Salt Creek Reservoir	--	
Hartsville 1	Hartsville, TN	1233	Old Hickory Reservoir	--	Steel primary containment
Hartsville 2	Hartsville, TN	1233	Old Hickory Reservoir	--	Steel primary containment
Skagit 1	Richland, WA	1277	Skagit River	--	Steel primary containment
Skagit 2	Richland, WA	1277	Ranney Wells	--	Steel primary containment
Total: 12 plants					

Sources: F. A. Heddleson, 1978, "Summary Data for U.S. Commercial Nuclear Power Plants," ORNL-NUREG-NSIG-141.

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APPENDIX E

LWR CONTAINMENT CHARACTERISTICS

The following table gives additional properties of the containments for all the U.S. nuclear plants. The plants are listed by type of containment and where available the design pressure, volume, and leak-rate are given, as well as seismic design standard and other comments.

LWR CONTAINMENT CHARACTERISTICS

1. DRY CONTAINMENTS (PWR UNLESS NOTED)

Plant name	Primary containment			Shield building	DBA earthquake acceleration (g)
	Design pressure (psig)	Volume (10^6 ft^3)	Leak rate (%/day)		
Yankee Rowe	31.5		3		
Big Rock Pt. (BWR)	27		0.5		0.05
San Onofre 1	46.4	1.44	0.5		0.50
Haddam Neck	40		0.1		0.17
Ginna 1	60		0.1		0.20
Indian Pt. 2	47		0.1		0.15
Turkey Pt. 3, 4	59	1.5	0.25		0.05
Palisades	55	1.64	0.2		0.20
Point Beach 1, 2	60		0.4		0.12
Oconee 1, 2, 3	59	1.9	0.5		0.05-0.15
Salem 1, 2	47		0.1		0.15
Diablo Canyon 1, 2	47		0.1		0.20
Prairie Island 1, 2	41.4		0.5	Yes	0.12
Ft. Calhoun 1	60	1.05	0.2		0.17
Indian Pt. 3	47	2.61	0,1		0.15
Three Mile Island 1	55	2.0	0.2		0.12
Zion 1, 2	47		0.1		0.17
Crystal River 3	55		0.25		0.10
Kewaunee	46		0.5	Yes	0.12
Maine Yankee	55		0.1		0.10
Rancho Seco	59		0.1		0.25
AK Nuclear 1	59		0.2		0.20
Calvert Cliffs 1,2	50		0.33		0.15

1. DRY CONTAINMENTS (PWR UNLESS NOTED) - CONTINUED

Plant name	Primary containment			Shield Building	DBA earthquake acceleration (g)
	Design pressure (psig)	Volume (10^6 ft^3)	Leak rate (%/Day)		
Midland 1, 2	67		0.1		0.10
Millstone 2	54		3		0.17
St. Lucie 1, 2	39.6	2.5	0.05		0.05
Trojan	60		0.2		0.25
Davis Besse 1	40		0.5	Yes	0.15
Farley 1, 2	54		0.3		0.10
AK Nuclear 2	?		0.1		0.20
Waterford 3	39.6		0.5	Yes	0.10
Summer 1	55		0.2		0.12
Harris 1, 2	42	2.5	0.3		0.12
LaCrosse (BWR)	52	0.26	0.1		?
Vogtle 1, 2	47	2.75	0.1		0.20
Bellfonte 1, 2	50	3.8	0.2		0.18
Seabrook 1, 2	48	2.7	0.2		0.20
Comanche Peak 1, 2	50	2.5	0.2		0.12
Byron 1, 2	50	2.9	0.1		0.12
Braidwood 1, 2	50	2.9	0.1		0.12
Washington 1	46.4	3.1	0.2		0.25
Wolf Creek	60	2.5	0.1		0.20
Callaway 1	60	2.5	0.1		0.20
Cherokee 1	49.5	3.3	0.2	Yes	0.15
South TX 1, 2	56.5	3.3	0.3		0.10
Washington 3	44	3.4	0.5	Yes	0.32
Palo Verde 1,2,3	60	2.7	0.1		0.20
Marble Hill 1, 2	50	2.9	0.1		0.12
Yellow Creek 1, 2	45	3.5	0.2	Yes	0.30

2. ICE CONDENSER (PWR)

Plant name	Primary containment			Shield building	DBA earthquake acceleration (g)
	Design pressure (psig)	Volume (10^6 ft^3)	Leak Rate (%/day)		
Cook 1,2	12		0.25		0.20
Sequoyah 1,2	10.8	1.2	0.5	Yes	0.14
Watts Bar 1,2	13.5		0.5	Yes	0.18
Catawba 1,2	15	1.2	0.2	Yes	0.15
McGuire 1,2	15		0.2	Yes	0.12

3. SUBATMOSPHERIC (PWR)

Plant name	Primary containment			Shield building	DBA earthquake acceleration (g)	Normal containment pressure (psia)
	Design pressure (psig)	Volume (10^6 ft^3)	Leak rate (%/day)			
Surry 1,2	45	1.8	0.1		0.15	9-11
North Anna 1,2	45		0.1		0.15	9-11
Beaver Valley 1,2	45	1.8	0.1		0.10	9-11
Millstone 3	45	2.3	0.25		0.17	9.5-11

4. MARK I (BWR)

Plant Name	Primary Containment				Secondary		DBA Earthquake Acceleration (g)	Suppression Pool Volume (10^6 ft^3)
	Wetwell Design Pressure (psig)	Drywell Design Pressure (psig)	Free Volume (10^6 ft^3)	Leak Rate (%/Day)	Design Pressure (psig)	Leak Rate (%/Day)		
Oyster Creek 1	35	62		0.5	.25	100	0.22	
Nine-Mile Pt 1	35	62		0.5		100	0.11	
Dresden 2, 3	62	62	.27	0.5	.25	100	0.20	0.11
Quad-Cities 1, 2	56	56		0.5	7" W.C.	4000 cfm	0.12	
Browns Ferry 1, 2, 3	56	56	.27	0.5	2" W.C.	100	0.20	0.085
Monticello	56	56	.24	0.5	.25	100	0.12	0.277
VT Yankee	56	56	.28	0.5	.25	100	0.14	0.078
Peach Bottom 2, 3	56	56	.28	0.5	.25	100	0.12	0.14
Pilgrim 1	56	56	.27	0.5	.25	100	0.15	0.084
Cooper	56	56	.25	0.5	.25	100	0.20	0.087
Hatch 1, 2	56	56	.26	1.2	.25	100	0.15	0.087
Arnold	56	56	.20	0.5	.25	100	0.12	0.059

4. MARK I (BWR) - CONTINUED

Plant Name	Primary Containment				Secondary			
	Wetwell Design Pressure (psig)	Drywell Design Pressure (psig)	Free Volume (10^6 ft^3)	Leak Rate (%/Day)	Design Pressure (psig)	Leak Rate (%/Day)	DBA Earthquake Acceleration (g)	Suppression Pool Volume (10^6 ft^3)
Fitzpatrick	56	56		0.5	.25	100	0.15	
Fermi 2	56	56	.29	0.5	.25	100	0.10	0.12
Hope Creek 1	56	56	.30	0.5	3	10	0.15	0.12
Brunswick 1, 2	53	53	.29	0.5	.25	100	0.16	0.087

5. MARK II (BWR)

Plant Name	Primary Containment				Secondary		DBA Earthquake Acceleration (g)	Suppression Pool Water Volume (10^6 ft^3)
	Wetwell Design Pressure (psig)	Drywell Design Pressure (psig)	Free Volume (10^6 ft^3)	Leak Rate (%/Day)	Design Pressure (psig)	Leak Rate (%/Day)		
Shoreham	56	56	0.33	0.5	.25	100	0.15	0.081
Limerick 1, 2	55	55	0.40	0.5	.25	100	0.12	0.12
Zimmer 1	45	45	0.27	0.5	.25	100	0.10	0.10
LaSalle 1, 2	45	45	0.39	0.5	.25	100	0.15	0.11
Susquehanna 1, 2	48	48	0.39	0.05	.25	100	0.10	0.12
Nine Mile Pt. 2	45	45	0.39	1.1	.25	100	0.10	0.14
Washington 2	40.5	40.5	0.35	0.5	.25	100	0.25	0.14

6. MARK III (BWR)

Plant Name	Primary Containment			Secondary		DBA Earthquake Acceleration (g)	Suppression Pool Water Volume (10^6 ft^3)
	Drywell Design Pressure (psig)	Free Volume 10^6 ft^3	Suppression Chamber Design Pressure (psig)	Design Pressure (psig)	Leak Rate (%/Day)		
Grand Gulf 1, 2	23	1.7	15	15	0.1	0.15	0.16
Perry 1, 2	23	1.2	15	.25" W.C.	100	0.15	0.15
River Bend 1, 2	25	1.4	15	-10" W.C.	1800	0.10	0.16
Clinton 1, 2	25	2.0	15	15		0.15	0.16
Harts-ville 1, 2	25	1.4	15	0.2	0.1	0.18	0.15
Skagit 1, 2	30	2.1	15	15	0.1	0.25	0.15

Source: F. A. Heddleson, "Design Data and Safety Features of Commercial Nuclear Power Plants", Vols I-V, ORNL- NSIC-55 & 96, 1975 & 1976.