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Application of Containment and Release Management Strategies to PWR Dry-Containment Plants

Prepared by
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Prepared for
U.S. Nuclear Regulatory Commission

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Abstract

This report identifies and evaluates accident management strategies that are potentially of value in maintaining containment integrity and controlling the release of radioactivity following a severe accident at a pressurized water reactor with large-dry containment. The strategies are identified using a logic tree structure leading from the safety objectives and safety functions, through the mechanisms that challenge these safety functions, to the strategies. The strategies are applied to severe accident sequences which have one or more of the following characteristics: significant probability of core damage, high consequences, give rise to a number of potential challenges, and include the failure of important safety systems. Zion and Surry are selected as the representative plants for the atmospheric and sub-atmospheric designs, respectively.

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Executive Summary

The purpose of the present report is to identify, as well as to assess, accident management strategies which could be important for preventing containment failure and/or mitigating the release of fission products during a severe accident in a PWR plant with a large dry containment. While the development of detailed actions is of necessity plant specific, the ideas contained in this report can be useful to individual licensees who are in the process of developing their accident management programs. The report should also be helpful to a reviewer of a licensee's accident management plan. Two types of containments are considered, atmospheric and sub-atmospheric. The Zion Nuclear Power Plant Unit 1 and the Surry Nuclear Power Station Unit 1 respectively, are used as the example plants in this report. Some of the variations among the other PWR large dry plants are also discussed.

The present report emphasizes the use of existing plant capabilities for severe accident management. The containment and release management (CRM) strategies differ from the existing emergency response guidelines (ERG) primarily in terms of the conditions under which certain actions are undertaken and certain systems activated. For CRM, systems are often operated in an anticipatory instead of a response mode, and often beyond their design limits. Non-safety grade systems are also made use of for CRM. The plant features that are important to containment and release management in a large dry containment are reviewed to identify their function and performance under severe accident conditions. These include the containment design as well as the plant systems and the resources needed to support their operation. Important issues related to these systems and some of the uncertainties involved in severe accident phenomena are discussed.

Maximum use was made of information contained in currently available safety studies related to PWR dry containments in general, and the Zion and Surry plants in particular.

As a result of the examination conducted in this study a safety objective tree was developed, which links the general safety objectives of containment and release management with the strategies identified as helpful in mitigating the challenges in a large dry PWR containment.

The strategies were assessed by application to certain accident sequences, i.e. Station Blackout sequences, LOCA sequences, and SGTR sequences. A DCH event was also singled out for discussion. The strategies discussed may, of course, also be of benefit in other sequences than the ones considered in this report.

Because of their large containment volume and high design pressure, PWR dry containments are relatively robust and provide considerable opportunities to maintain containment integrity and minimize the release of radiation following a severe accident.

Among the combustion modes which could potentially occur in such a containment, the most challenging appears to be a local detonation caused by non-uniform gas distribution, since this could threaten the containment integrity. The potential for local detonation depends on the containment construction, interior layout and other specific design parameters. However for large dry containments, the risk from all combustion modes is deemed low enough that no modification of these plants is necessary, although licensees should be cognizant of the potential for these events to occur. Plant specific combustion control should focus on promoting gas mixing and deliberate burning in order to keep the combustible gas concentration below the lean detonation limit.

The direct containment heating (DCH) associated with a high-pressure melt ejection (HPME) event appears to be an early threat to containment integrity. Many factors can influence the effects of melt ejection and some are not well enough understood to allow unequivocal statements regarding their influence on DCH. For instance, the impact of co-dispersal of water present in the reactor cavity during a HPME event involves considerable uncertainty. However, mitigation or elimination of the DCH effect could be accomplished by RCS depressurization.

Overpressurization of PWR containments can occur during the late phase of an accident due to the buildup of steam and noncondensible gases. However, because of the large containment volume, for most PWR plants overpressurization is a slow process. In most cases, the ultimate capacity of the containment would not be reached for a number of days. Under these circumstances, mitigation can be achieved by the restoration of containment cooling systems, using alternate water sources, or by a controlled venting. Restoration of containment cooling systems must be done cautiously so as not to de-inert the atmosphere and cause a sudden burn of a large quantity of combustible gases, which may have accumulated in the containment.

Basemat melt-through also is a potentially important challenge during the late phase of the accident for some containment designs. However, concrete erosion by the molten core debris is a very slow process and it would take days for the concrete to lose its structural integrity. The erosion of the concrete may be mitigated by flooding the reactor cavity. However, there is a large uncertainty regarding the effectiveness of cavity flooding since it depends on the cavity configuration and the state of the core debris in the cavity.

For some PWRs, containment bypass events provide a significant contribution to the risk estimates. The mitigation strategies that are discussed in this report include the isolation of the break line, reactor coolant systems (RCS) depressurization, refilling of the refueling water storage tank (RWST), flooding the break location, and the activation of auxiliary building fire sprays. These strategies are currently feasible for many PWRs. However, for some plants, modification of existing systems and/or procedures are required.

The decision to carry out a strategy during a severe accident, depends on balancing the potential adverse consequences of strategy implementation against the consequences that could result if the strategy is not implemented.

List of Acronyms

AB	Auxiliary Building
AOP	Abnormal Operating Procedure
AFWS	Auxiliary Feed Water System
AVS	Annulus Ventilation System
BERG	Beyond Emergency Response Guidelines
BWR	Boiling Water Reactor
CCWS	Component Cooling Water System
CDF	Core Damage Frequency
CE	Combustion Engineering
CHRS	Containment Heat Removal System
CLWG	Containment Loads Working Group
CPWG	Containment Performance Working Group
CRM	Containment and Release Management
CSF	Critical Safety Function
CSS	Containment Spray System
CST	Condensate Storage Tank
DCH	Direct Containment Heating
DBA	Design Basis Accident
DIS	Distributed Igniter System
ECA	Emergency Contingency Action
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
ERCWS	Emergency Raw Cooling Water System
ERG	Emergency Response Guideline
ESF	Engineered Safety Features
FP	Fission Products
FRG	Function Restoration Guideline
HEPA	High Efficiency Particulate Air
HPIS	High Pressure Injection System
HPME	High Pressure Melt Ejection
HSS	Hydrogen Sampling System
IDCOR	Industry Degraded Core Rulemaking
IPE	Individual Plant Examination
ISLOCA	Interfacing Systems LOCA
LOCA	Loss of Coolant Accident
LPIS	Low Pressure Injection System
LRC	Lower Reactor Cavity
LWR	Light Water Reactor
MSIV	Main Steam Isolation Valve
MWR	Metal Water Reaction
NSSS	Nuclear Steam Supply System
PDS	Plant Damage State
PLOCAP	Post LOCA Protection System
PORV	Power Operated Relief Valve
PSA	Probabilistic Safety Analysis
PSS	Probabilistic Safety Studies
PWR	Pressurized Water Reactor
RCFCS	Reactor Containment Fan Cooler System
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHRS	Residual Heat Removal System
RPV	Reactor Pressure Vessel
RWST	Refueling Water Storage Tank

List of Acronyms

SAM	Severe Accident Management
SBO	Station Blackout
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SOT	Safety Objective Tree
SWS	Service Water System
TSC	Technical Support Center
VB	Vessel Breach
WOG	Westinghouse Owner's Group

1 Introduction

1.1 Background

Experience obtained from Probabilistic Risk Assessment analyses indicates that a cost effective means for licensees to reduce severe accident risk even further is to supplement plant operating procedures with additional guidance for severe accidents, that is, by preplanned management of severe accidents. While minor hardware modifications may in some cases be necessary to implement the resulting procedural changes or additions, much can be accomplished through innovative use of already existing plant systems. Such an approach to risk reduction is preferable to one which relies on significant, and therefore costly, hardware changes or additions.

The current phase of the NRC Research effort in identifying and assessing accident management actions is concerned with mitigative strategies which would most likely be applied in the more advanced stages of a severe accident. Before vessel failure the emphasis is on arresting or mitigating core damage progression in the reactor vessel. If vessel failure has already occurred or is imminent the emphasis is on maintaining containment integrity, quenching core debris ex-vessel, and minimizing fission product release to the environment. Containment and Release Management (CRM) constitutes the mitigative aspects of Severe Accident Management. CRM anticipates a breach in the reactor coolant system pressure boundary and, through the effective, innovative and informed use of available systems, seeks to maintain containment integrity and minimize radioactive release following a severe accident.

Brookhaven National Laboratory is producing a series of reports dealing with the containment and release management part of a severe accident. The mitigative strategies discussed in these reports are often applied in situations where present understanding of the phenomena encountered is limited. Therefore, the uncertainty surrounding some of these strategies is quite large. Also, many of the suggested strategies go well beyond existing procedures. Often the strategies and the challenges which they address depend on the specific containment types and therefore five individual reports are being published for containment and release management, each one addressing the challenges and strategies applicable to one of the five containment types used in the U.S. today. The present report is one of this series.

1.2 Objective, Scope and Approach

The purpose of the present report is to identify, as well as to assess, accident management strategies which could be important for preventing or delaying containment failure and/or mitigating the release of fission products during a severe accident in a PWR plant using a large dry containment. The discussions contained in this report are intended to provide useful information to licensees formulating a severe accident management plan for their individual plants. While the development of detailed guidance is of necessity plant specific, the ideas contained in this report can be useful to individual licensees who are in the process of developing such guidance.

The report can also furnish the reviewer of an accident management plan with a systematic overview of the challenges a PWR with a large dry containment may face during a severe accident and the strategies which could be used to meet these challenges.

In the sections which follow the challenges that can impair containment integrity and give rise to fission product releases from a large dry containment during a severe accident are discussed. Strategies which can be used to eliminate or mitigate the effect of some of these challenges are identified. That is, actions in the form of accident management strategies are identified where appropriate and possible, and their anticipated effect on the accident is assessed. Not all challenges can be completely met by available strategies.

Strategy identification can be enhanced and summarized via a safety objective tree (SOT). A tree structure was developed to link the appropriate safety objectives with the challenges of the accident and ultimately with the strategies devised to meet these challenges. This tree structure is similar to that used in previous accident management reports.

For containment and release management two safety objectives apply: (1) preventing containment failure, and (2) mitigating fission product release to the environment. These safety objectives are achieved by the maintenance of certain safety functions. During an accident the normal operation of the safety functions will be threatened by particular challenges which arise from a variety of mechanisms that can occur in the plant. These mechanisms can in turn be prevented or mitigated by a number of strategies. The tree developed by this process for a large dry containment is illustrated in Figure 3.1.

The systematic method used in this report for strategy identification and the top down structure of the SOT, using the hierarchy just described, allow an analyst to decompose the problem of strategy identification into more and more detailed levels in an organized manner. This systematic method of challenge depiction and strategy identification is more likely to achieve a certain degree of completeness than other more haphazard identification processes.

Previous history of the accident often plays an important role in determining which strategies should be implemented and how successful their implementation will be. To account for these factors certain accident sequences are selected and the strategies are assessed in the context of these sequences. However, the identified strategies are not only applicable to the sequences discussed. The strategies will often be beneficial under other conditions as well, although these conditions may need to be accounted for in strategy implementation.

1.3 Organization of the Report

The subsequent sections of the report are arranged as follows. Section 2 describes the PWR dry containment system, and the plant safety systems and resources, as well as existing severe accident management capabilities. A detailed examination of the containment challenges and the identification of the relevant containment and release strategies for a PWR large dry plant are presented in Section 3. The application of the strategies during certain accident sequences is discussed in Section 4. Section 5 summarizes the important findings. References are contained in Section 6.

2 Plant Capabilities

2.1 The PWR Dry Containment System

2.1.1 Characteristics of PWR Dry Containments

Three types of construction techniques have been used for currently existing PWR dry containments: 1) reinforced concrete, 2) prestressed concrete, and 3) steel.

A reinforced concrete containment has three basic structural elements, namely, the basemat, cylinder, and dome. Reinforcing bars are placed in all three elements. The containment accomodates the design basis loads via the reinforced concrete and through the net free volume of the containment. Many reinforced concrete containments have a steel liner attached to, and supported by, the concrete. The liner functions primarily as a gas-tight membrane and also transmits loads to the concrete. All subatmospheric plants, such as Surry, use reinforced concrete as shown in Figure 2.1.

In more recent plants the reinforced concrete design has been replaced, to a large extent, by fully prestressed containments. In this design, the reactor containment is in the shape of a cylinder with a shallow domed roof and a flat foundation slab. The cylindrical portion is prestressed by a post-tensioning system consisting of horizontal and vertical tendons. The dome has a three-way post-tensioning system. The foundation slab is conventionally reinforced with high-strength reinforcing steel. The entire structure is lined with a one-quarter inch welded steel plate to provide vapor tightness. A prestressed concrete containment requires less net free volume for a given blowdown load. The external force applied by the tendons allows a higher internal pressure. Zion, shown in Figure 2.2, is a representative plant for this category.

Most steel containments utilize a steel plate interior structure enclosed by a separate biological shield concrete building. (Exceptions are San Onofre 1 and Yankee-Rowe which lack the concrete shield building.) The concrete shield structure is not designed for high internal pressure but serves to protect the steel shell from extreme environmental effects. The internal pressure of the containment is carried by the structural strength of the steel plating. A typical steel shell design, Davis Besse 1, is shown in Figure 2.3.

An important feature of a PWR dry containment which can significantly influence the progression of a severe accident is the configuration of the reactor cavity located below the reactor vessel. There is a large variation in reactor cavity design among PWR dry containment plants. This is due, in part, to the fact that the cavity plays no role in design basis accidents. However, under severe accident conditions, the reactor cavity could strongly affect the challenges imposed on the containment. The cavity's size, geometry and outlets into the containment regions could affect the interactions of corium/water and corium/concrete, and these in turn could affect the subsequent containment pressurization and basemat erosion. For example, the presence or absence of sumps or curbs around access ports to the cavity region would determine whether the cavity would be flooded during a particular accident sequence, thereby influencing whether the core debris could be cooled. The outlet flow paths from the cavity can significantly impact the amount of material which might be ejected into other containment regions by a high pressure release from the reactor vessel. The Industry Degraded Core Rulemaking (IDCOR) program has classified PWR reactor cavities into 14 types according to geometry to express their expectations of debris dispersal during a high-pressure melt ejection accident [1].

The Zion and Surry plants were analyzed as part of the NRC's NUREG-1150 program [2], while the Industry Degraded Core Rulemaking (IDCOR) program studied the Zion plant. Both plants will be referred to extensively during discussions presented in this report. The results obtained for these two plants can be used [illegible]tatively as a guide for the evaluation of other PWR plants with a large dry containment design. However, due to some unique characteristics of Zion and Surry, such as the reactor cavity geometry, not all the insights of this report may be valid for other plants.

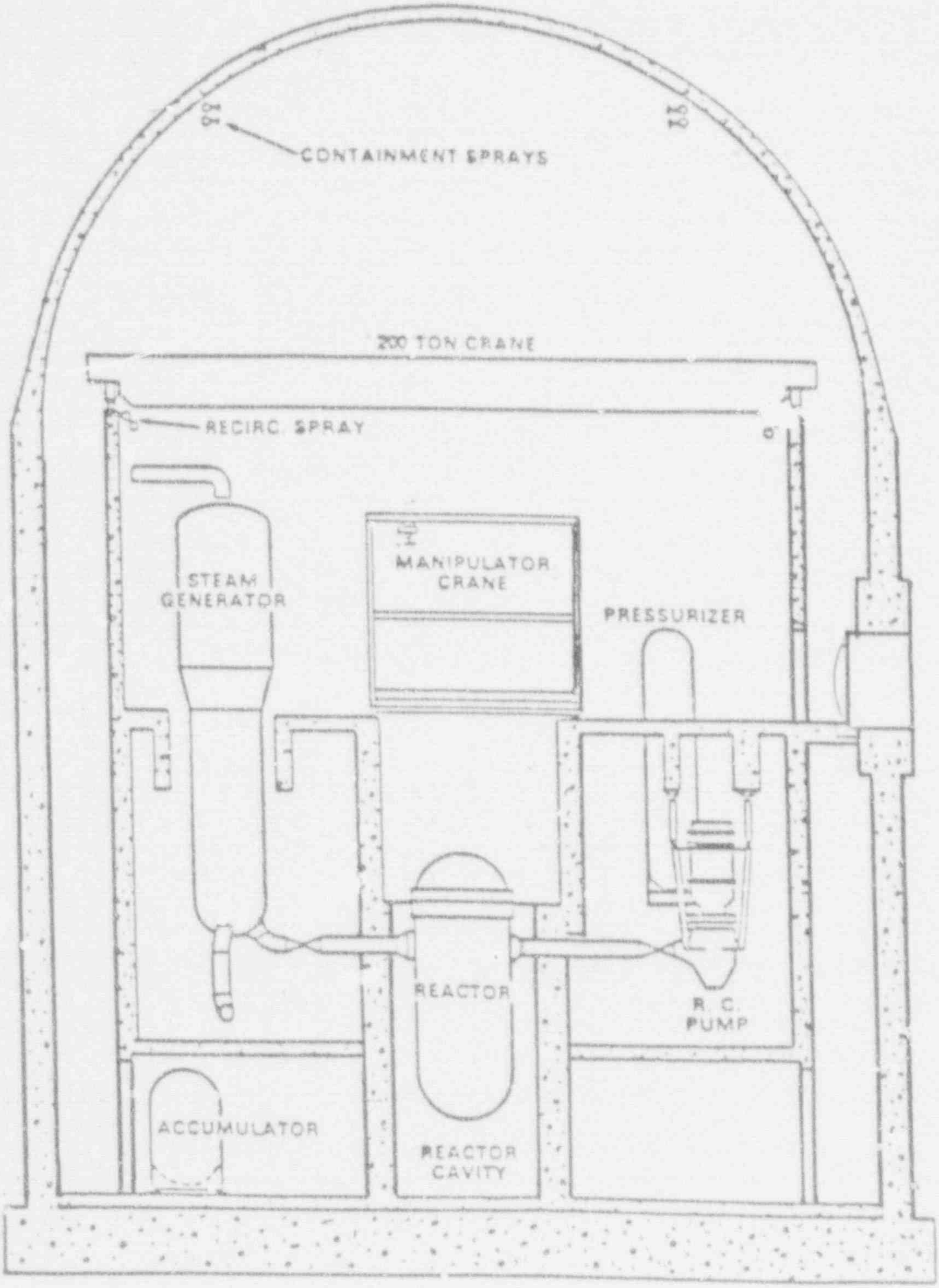


Figure 2.1 Reinforced Concrete Containment (Surry)

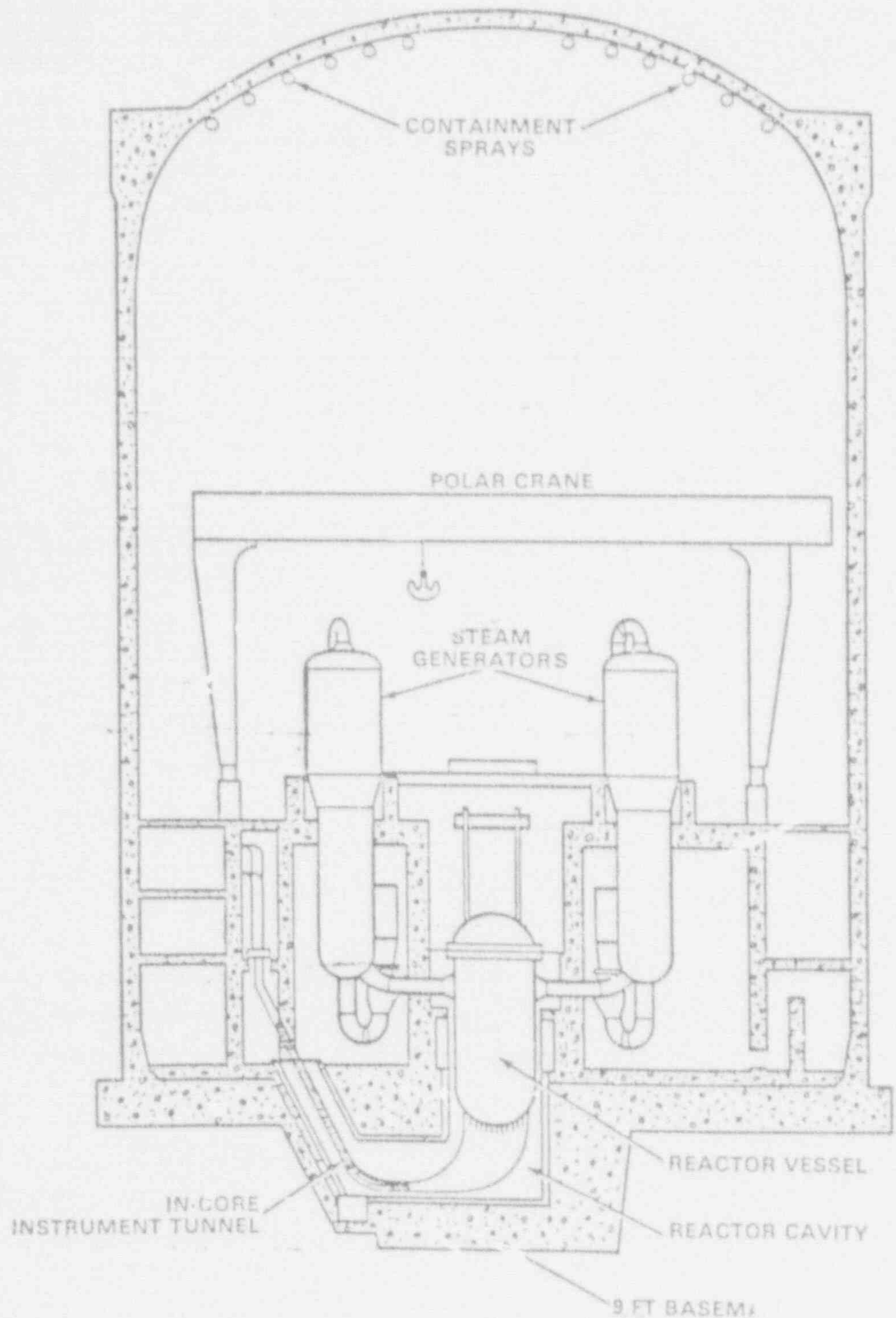


Figure 2.2 Prestressed Concrete Containment (Zion)

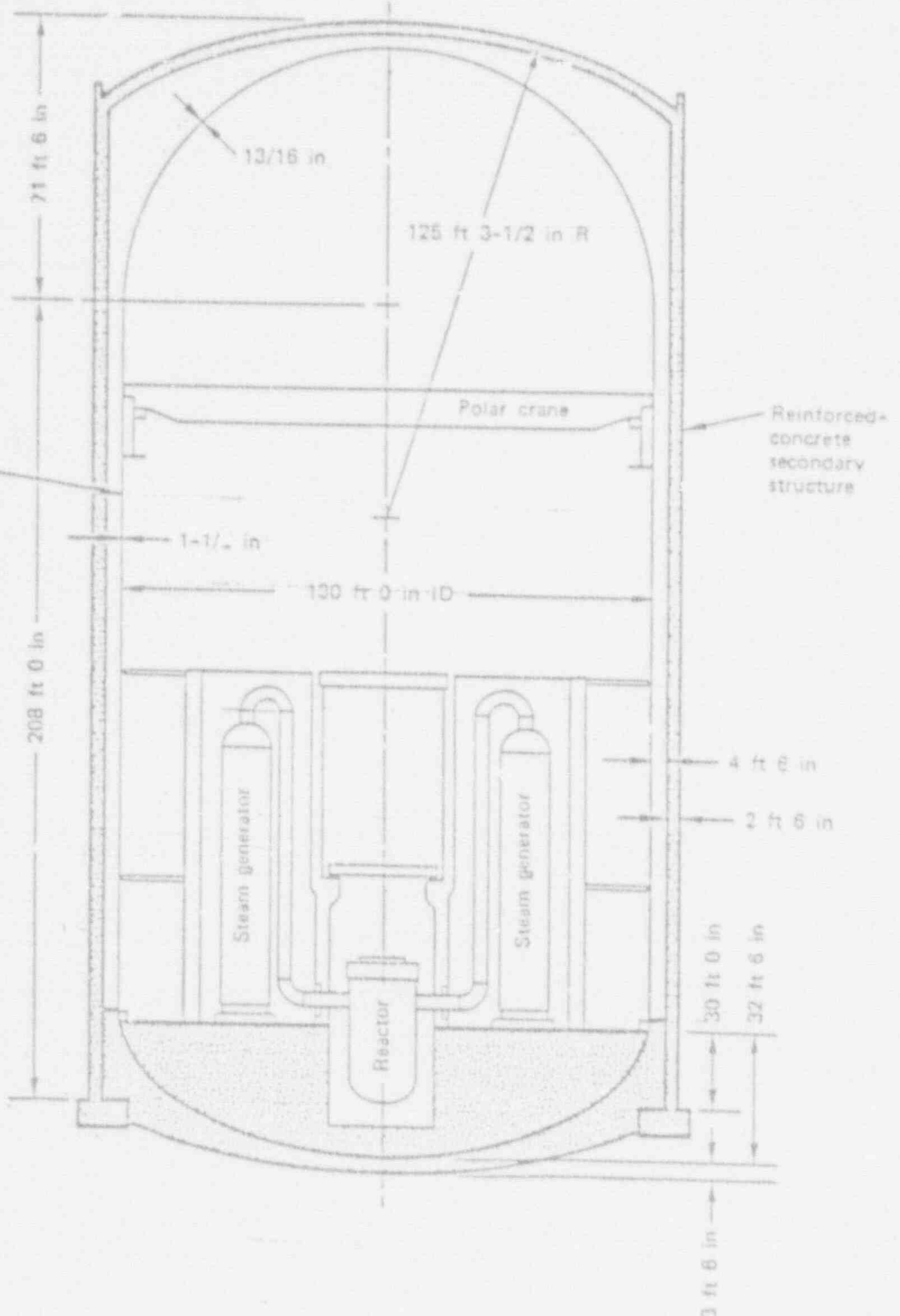


Figure 2.3 Steel Dry Containment With Reinforced Concrete Secondary Containment (Davis Besse 1)

2.1.2 Containment Pressure Capability and Failure Mode

The design pressures of PWR dry containments vary between 25 to 60 psig. Blejwas [3] has correlated variations in the design pressure with the various types of containment structures as given below:

	Atmospheric Design Pressure (psig)	Subatmospheric Design Pressure (psig)
A. Concrete		
a. Prestressed vertical cylinder and dome with flat base	47-60	--
b. Reinforced hemispheric dome, vertical cylinder and flat base	42-55	45
c. Others	42-60	--
B. Steel		
a. Hemispheric dome, cylindrical body, and ellipsoidal base	34-44	--
b. Sphere	25-48	--

The higher design pressures are associated with smaller prestressed concrete containments; the lower pressures with non-stress-relieved steel structures; and the mid-range pressures with reinforced concrete containments, and stress-relieved steel structures. The design pressure can be considered as a measure of containment capability under severe accident conditions. For plants whose ultimate capacity is not determined through detailed strength/stress analysis, the containment failure pressure can be estimated as being 2 to 3 times the design pressure.

The potential failure pressures for the Zion and Surry plants from aggregating of expert-specific probability distributions provided for the NUREG-1150 study [2] are given below:

	Zion	Surry
Mean Failure Pressure, psig	134	128
Failure pressure at 5th - 95th percentile range, psig	108-180	95-150

Potential failure modes that could cause early containment failure are:

1) Overpressurization due to direct containment heating of the containment atmosphere by the core debris ejected from the reactor vessel at high pressure,

2) Early combustion of hydrogen generated during the core degradation process, and

3) Steam spike generated at the time of vessel breach due to core debris/water interaction in a flooded reactor cavity.

Late containment failure could be caused by:

1) Basemat meltthrough,

2) Overpressurization due to noncondensible gases and steam generated by the interaction of the core debris with concrete and water, and

3) Late hydrogen combustion.

In addition to the above failure modes, there is also the potential for containment bypass. Bypass sequences include steam generator tube rupture events and interfacing systems LOCA (ISL). Interfacing systems LOCA refers to accidents in which the interface between the high pressure reactor coolant system (isolation valves) and a low pressure secondary system is breached. If this occurs, the low pressure system will be overpressurized and could fail outside the primary containment. This failure would establish a flow path directly from the damaged core to the environment or to an intermediate, but low capacity, building.

The Zion and Surry containment analyses in NUREG-1150 made extensive use of expert judgement to quantify the containment event trees and estimate the probability of containment failure. The mean frequency of core damage due to internal events was predicted to be 3.4E-4 per year for Zion [2]. The frequency weighted average conditional probabilities of four accident progression bins are:

No containment failure	0.73
Late containment failure	0.24
Early containment failure	0.02
Containment bypass	0.01

Late containment failure is mainly from basemat meltthrough, and early containment failure is from a combination of in-vessel steam explosions, overpressurization and containment isolation failure.

For the Surry plant, the mean frequency of core damage due to internal events is about 4.1E-5 per year [2]. The frequency weighted average conditional probabilities of four accident progression bins are:

No containment failure	0.81
Late containment failure	0.06
Early containment failure	0.01
Containment bypass	0.12

2.1.3 Containment Fission Product Retention

The fission products generated during a core meltdown accident can be conveniently grouped as arising from two sources; one source is the release resulting from the degradation of the core materials in the reactor pressure vessel. Fission product groups, such as noble gases, Cs, I, and Te, are released before or at the time of vessel breach. Another release is caused by core-concrete interactions after vessel breach. Fission product groups represented by Te, Sr, Ru, La, Ba, and Ce are released mainly during the early time period when the cavity concrete floor is thermally attacked by the molten debris.

In addition to the natural deposition processes, which are related to the ratio of the deposition area and the containment volume, the retention of fission product in PWR dry containment is strongly affected by the containment spray system and the cavity configuration. Sprays are an effective means for removing airborne radioactive aerosols. Other than the release of noble gases and some iodine evolution, the release of radioactive material to the environment should be very small if sprays have operated for an extended time.

Many PWR plants, have a cavity configuration which allows the overflow of water from the containment floor into the cavity. However, this is not the case for all plants. For example, Zion has a curb as well as the "dog house" enclosure around the cavity which may limit water access. There may be other plants with curbs or enclosures. The presence of an overlaying pool of water on the core debris would mitigate the release of radionuclides from core-concrete interactions. Water in the cavity may also mitigate the release of radioactive materials from the molten core-concrete interactions if a coolable debris bed is formed. For PWR plants, such as Surry, which do not have a flow path for water to enter the cavity from the containment floor, the cavity will be dry at the time of vessel breach unless the containment spray system has operated. A dry-cavity will not be able to mitigate the release of radioactive materials from the molten core-concrete interaction.

2.2 Containment Safety Systems and Resources

2.2.1 Containment Heat Removal Systems

The high pressures and temperatures during an accident in a PWR dry containment may be reduced by two containment heat removal systems: the containment water sprays and the atmospheric fan coolers. In some designs both systems are Engineered Safety Features (ESFs) and are designed to operate during a LOCA assuming a single component failure. In other designs, typically the subatmospheric plants, only the sprays are an ESF system. The containment heat removal is accomplished by heat exchangers in the containment spray system and containment fan coolers. Typically, the sprays and fan cooler systems are sized to accommodate energies associated with the reactor decay heat and the sensible and latent heat of the primary system coolant.

The containment spray system (CSS) of a PWR lar[illegible] plant like Zion has three independent 100% capacity subsystems with no common headers. A single act[illegible] passive failure in any of these subsystems will not affect the operation of either of the other two subsystems. Of the three containment spray pumps, two are motor-driven and the third is diesel engine-driven. The capacity of each pump is 3000 gpm (Zion). During a LOCA event, a high-high containment pressure signal coincident with a safety injection signal will start all three containment spray pumps and open the normally closed motor operated valves on the discharge of these pumps. All three pumps take suction from the Refueling Water Storage Tank (RWST). When spray is required during the recirculation phase of the accident, two of the three spray subsystems can be supplied with water from the containment sump via the residual heat removal (RHR) pumps. Therefore, spray pump operation is not necessary during the recirculation phase. In a subatmospheric containment, like Surry, there typically are two spray systems; an injection spray system that functions as described above, and a recirculation spray system. When the RWST has been emptied in this plant, the injection spray system is secured and the recirculation spray system is started.

The Reactor Containment Fan Cooler System (RCFCS) is designed to filter, cool, and dehumidify the reactor containment environment during both normal and abnormal conditions. It is a recirculation system. There are a total of five units operating in parallel (Zion). For post accident operation a minimum of three units must function to satisfy safeguards requirements. Each RCFCS unit is composed of fan, motor, cooling coil, filters (both roughing filter assembly and HEPA filter assembly), moisture separator and backdraft dampers. For the Zion plant, the fan capacity during post accident conditions is 53000 cfm, and each unit is capable of removing an actual heat load of 81 x 10^6 Btu/hr and of achieving a steam condensation rate of 200 gpm.

A simple diagram of these containment cooling system heat transport paths is shown in Figure 2.4 [4]. The heat transfer path from the containment spray (or RHR) heat exchangers and the containment fan coolers to the ultimate heat sink is completed by one or two cooling water loops (i.e., the CCWS and/or the service water system).

2.2.2 Combustible Gas Control System

All PWR dry containments are equipped with Combustible Gas Control Systems in order to maintain the post-accident hydrogen buildup to a level below the flammability limit. The system contains four elements: 1) a hydrogen sampling system which alerts the plant operator to the hydrogen concentration in the containment, 2) a hydrogen/air mixing system which minimizes the formation of locally high H_2 concentrations, 3) hydrogen recombiners in which gases drawn from the containment are heated to high temperatures to combine hydrogen with oxygen and returned back to the containment, and 4) a containment purge system which allows venting of the containment atmosphere to the outside environment. However, these systems are designed to accommodate hydrogen accumulation for design

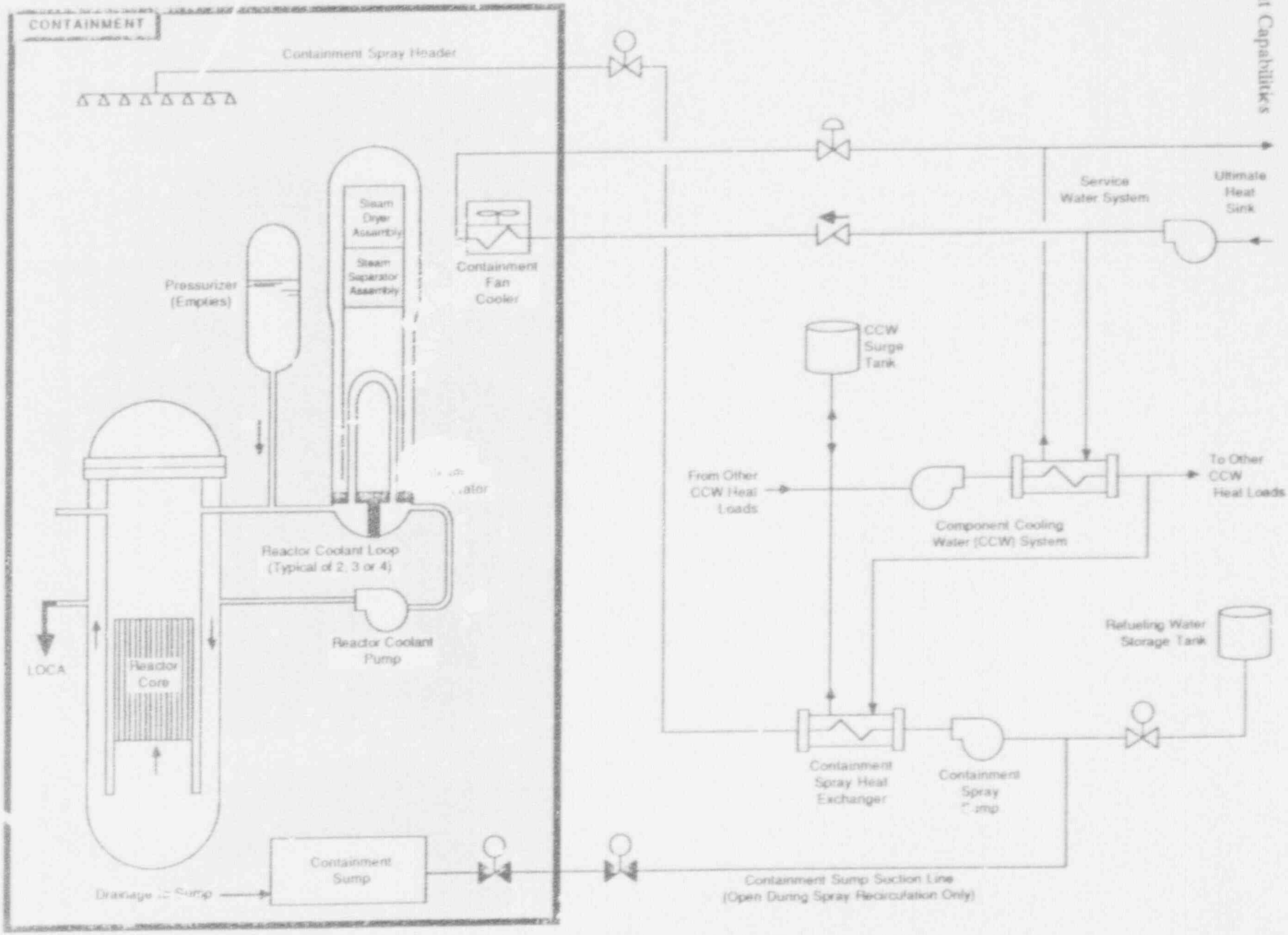


Figure 2.4 Typical PWR Containment Post-LOCA Heat Removal Paths

basis events (oxidation of 5% of the Zircaloy surrounding the active fuel). The systems are not designed for the hydrogen generation that might accompany a core meltdown accident.

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 the reaction of steam with fuel clad and steel structures, and the radiolytic decomposition of water. The accumulation of the hydrogen gas presents a threat to containment integrity if it is a sizeable quantity and if it were to burn rapidly or detonate. PWR dry containments are not required to have the intentional ignition systems that are required for the ice-condenser plants.

2.2.3 Containment Penetration and Isolation Systems

Containment penetration and isolation systems are designed to limit releases of radioactive gases and particulates to the environment after an accident. During normal operation, PWR containments are closed or have only a limited amount of purge flow. Following a LOCA, the containment isolation system is required to close isolation valves for nonsafety-related fluid systems penetrating the containment. The criteria defining the number and location of containment isolation valves in each fluid system depend on the function of the system and whether it is open or closed to the containment, atmosphere, or reactor system. Lines serving ESF systems remain in service subsequent to design basis events.

2.2.4 Component Cooling System

The purpose of the Component Cooling System (CCS) is to remove heat from systems which may contain radioactive water. The heat removal is then transferred to the Essential Raw Cooling Water System (ERCWS) for release to the environment. The CCS is a closed system, so it acts as a barrier between radioactive systems and the environment. The CCS is operated at lower pressures than any system with which it interfaces, so it will collect leakage. A PWR CCS typically consists of five pumps, four thermal barrier booster pumps, three heat exchangers, two surge tanks, and a CCS pump seal water collection unit. The systems served by the CCS include the containment spray system and residual heat removal system. An example of component cooling system heat transport paths is shown in Figure 2.5 [4].

2.2.5 Service Water System

The Service Water System (SWS) is also called Essential Raw Cooling Water System (ERCW) and is used during all phases of operation to remove heat from engineered and non-engineered safety systems. It is an open system, using water from the environment and discharged back to the environment.

The Zion plant has six pumps feeding two separate main supply headers, one header for each unit and three pumps on each header. The headers are crosstied so that any combination of pumps can serve both units under normal operating conditions. The system pressure is maintained at 55-75 psig in the main supply header and the pumps are rated at 22,000 gpm.

The service water system could be considered as an alternate water source to provide a long-term supply of water for the removal of heat from the containment. This strategy could be implemented during an accident in which all higher priority water supplies and systems are unavailable or inadequate. The strategy is accomplished by providing backup emergency connections, such as service water supply to RWST, CSS, or to a reactor cavity flooding system, if such a system exists. Actual hard-piped crossties between systems needed to implement this strategy are unlikely to exist in most plants. The alternate would be to utilize a temporary hose connection arrangement. Although such an arrangement would depend on specific plant configuration, it is likely that some plants have a penetration or blank flange that could be modified to permit a hose connection.

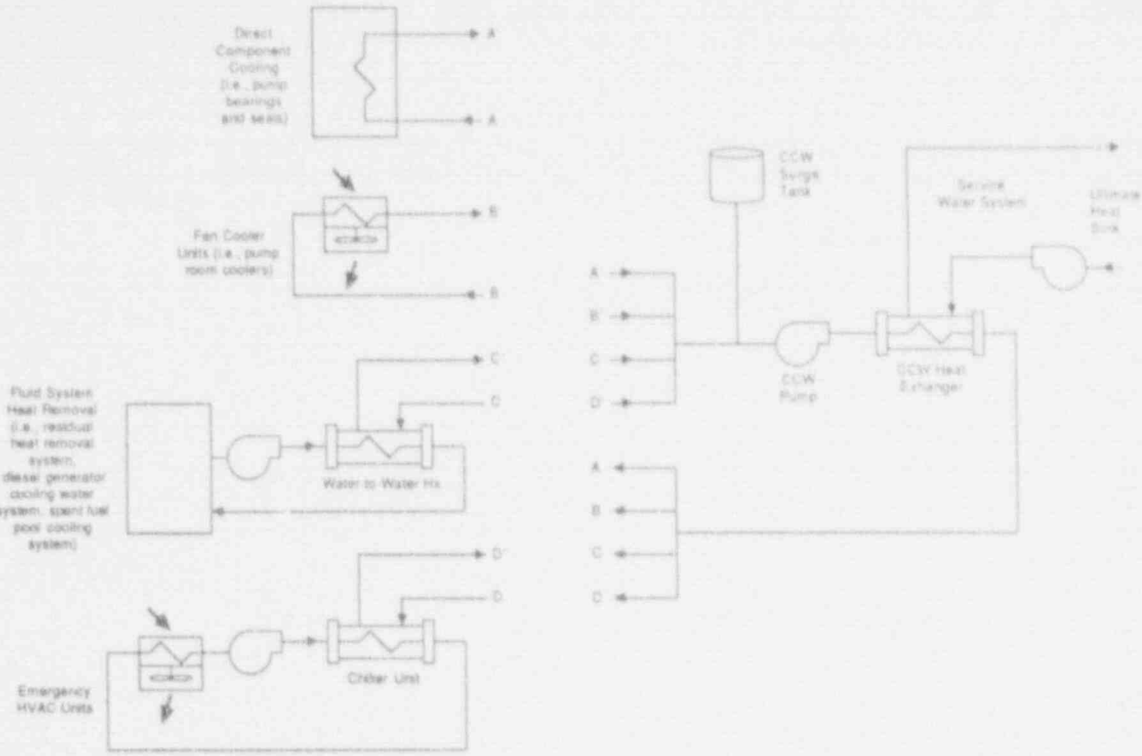


Figure 2.5 Examples of Component Cooling System Heat Transport Paths [4]

2.2.6 Fire Water System

Similar to the Service Water System, the fire water systems could be considered an alternate water source for long-term cooling of the containment. The system is normally maintained at 125 psig to 140 psig by the service water booster pumps which take their suction from the service water system (Zion). There are two fire pumps; one motor driven pump supplied with electrical power from the essential bus, and the other a diesel engine driven pump started by battery. Both of the fire pumps are rated at 2000 gpm. When the pressure in the fire protection system header falls to 115 psig, the motor driven fire pump will automatically start and restore the pressure to between 125 psig and 140 psig. If the motor driven pump fails to start or cannot meet the demand on the fire system, the diesel driven pump will automatically start at a header pressure of 110 psig.

Since the fire water system pressure is about twice as high as that of the service water system, it could be used when higher pressures are required. For CRM applications FW also requires the backup emergency connections.

3 Containment and Release Management Strategies

The development of Containment and Release Management (CRM) strategies requires an interface with the procedures, programs and policies which have already been developed for the Emergency Operating Procedures (EOPs) and the Emergency Response Guidelines (ERGs). The EOPs provide procedural direction for a wide range of adverse plant conditions during off-normal/abnormal events. The ERGs provide additional organizational structure and technical capabilities and on-site and off-site communications improvements to support the plant operating staff during an accident. The relationship between CRM strategies, EOPs and ERGs was discussed in References 5 and 6. In general, the EOPs are predominantly intended for use in the Control Room and constitute "early action", whereas CRM strategies are primarily intended for the Technical Support Center (TSC), or equivalent, and will constitute "late action". The ERGs provide "entry points" in the development of CRM strategies in the sense that they help to define the conditions of the reactor core, the reactor coolant system and the containment, as well as the status of safety systems. The strategies developed for CRM must be applicable to a wide range of possible conditions, and must also acknowledge the variety of phenomenological challenges which could be present.

3.1 Strategy Identification

The safety functions are used in EOPs and ERGs to prioritize actions and identify important equipment. This approach can be used to identify the CRM strategies. The safety function-based approach defines the relationship between the safety objectives of accident management; the safety functions needed to preserve these objectives; the challenges to the safety functions; the mechanisms causing these challenges; and the strategies which counter these mechanisms and thus mitigate or eliminate the effects of the challenges.

A schematic diagram of the safety objective tree for PWR plants with a dry containment is shown in Figure 3.1. The principal safety objectives of CRM are preserving containment integrity and minimizing the off-site release. If containment integrity is preserved little or no fission products are released. However, since containment integrity may be violated, not only by a bypass or failure of the containment, but also by a venting strategy intended to prevent unconditional failure, it becomes important to minimize the amount of fission products released under these circumstances.

The safety functions to prevent containment failure involve the control of containment isolation, pressure, and temperature. The challenges to containment failure consist of containment isolation failure, bypass, pressurization, and a severe thermal-environment. The safety function to mitigate off-site release involves fission product release control. The challenges are the release, transport and generation of fission products in the containment, and the release to the auxiliary building and the environment. The mechanisms which cause the containment challenges involve phenomenological processes which could potentially occur during accidents, and are discussed in detail in Section 3.2. The strategies to eliminate or mitigate these challenges are given in Section 3.3.

3.2 Containment Challenges

PWR dry containments are designed to accept the internal pressure resulting from design basis accidents (DBAs). However, the containment may be vulnerable to pressure and temperature loads associated with some low frequency core meltdown accidents which are potentially more severe than DBAs. Discussions of PWR containment challenges were given in the NUREG-1150 report [2] and in Reference 7. The containment challenges applicable to PWR CRM strategies are summarized below.

3.2.1 Containment Bypass

Accident Sequences that involve bypass of the containment were assessed to be important at the Zion and Surry plants, and were risk dominant at the Surry plant. The principal contributors to containment bypass are accidents initiated by interfacing-system loss-of-coolant (ISLOCA) and steam generator tube ruptures (SGTR). The predicted frequency of these events is quite small. However, the consequence of these events is high because their very occurrence implies immediate loss of containment integrity.

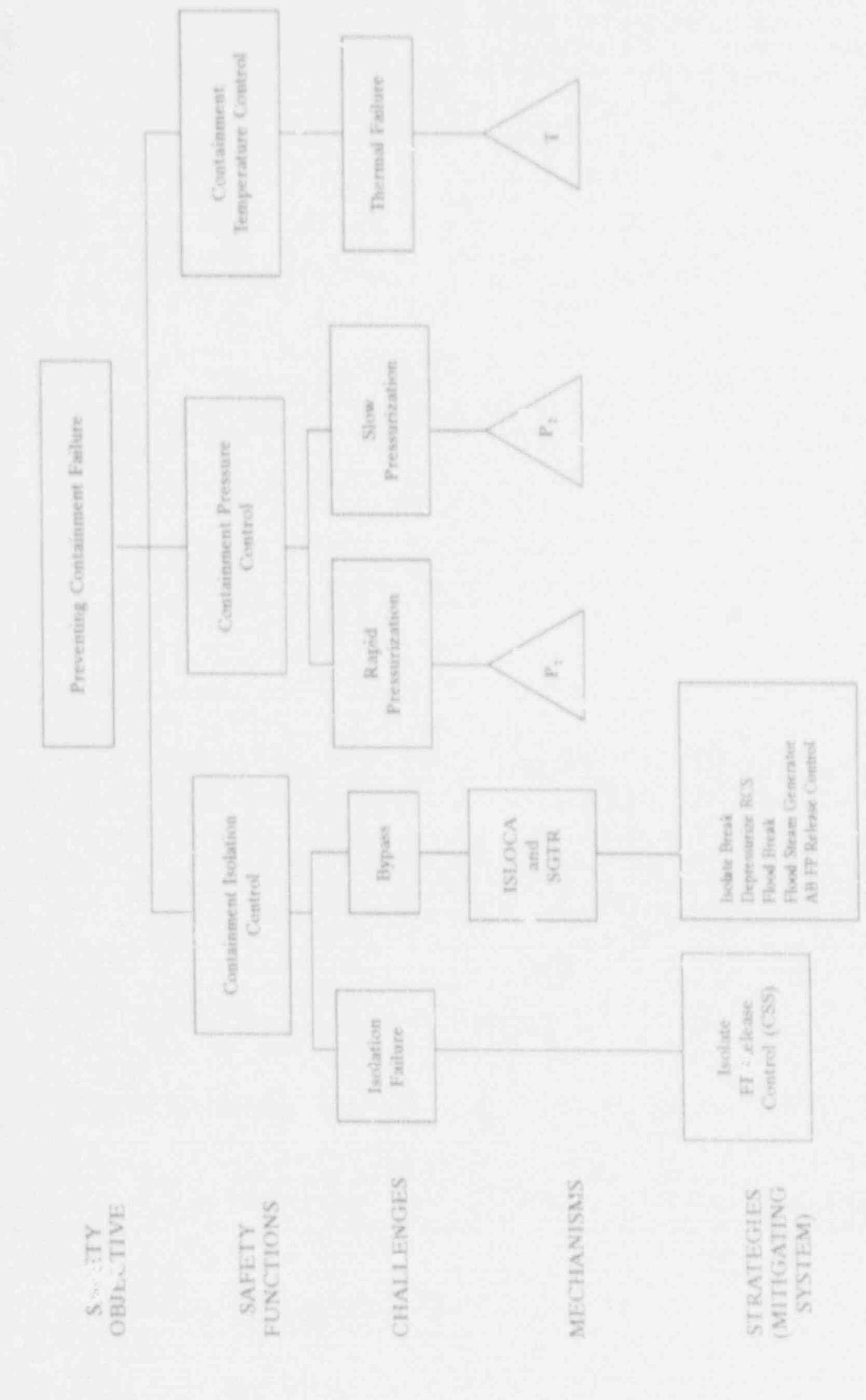


Figure 3.1 Safety Objective Tree for PWR Plants With Dry Containment Design

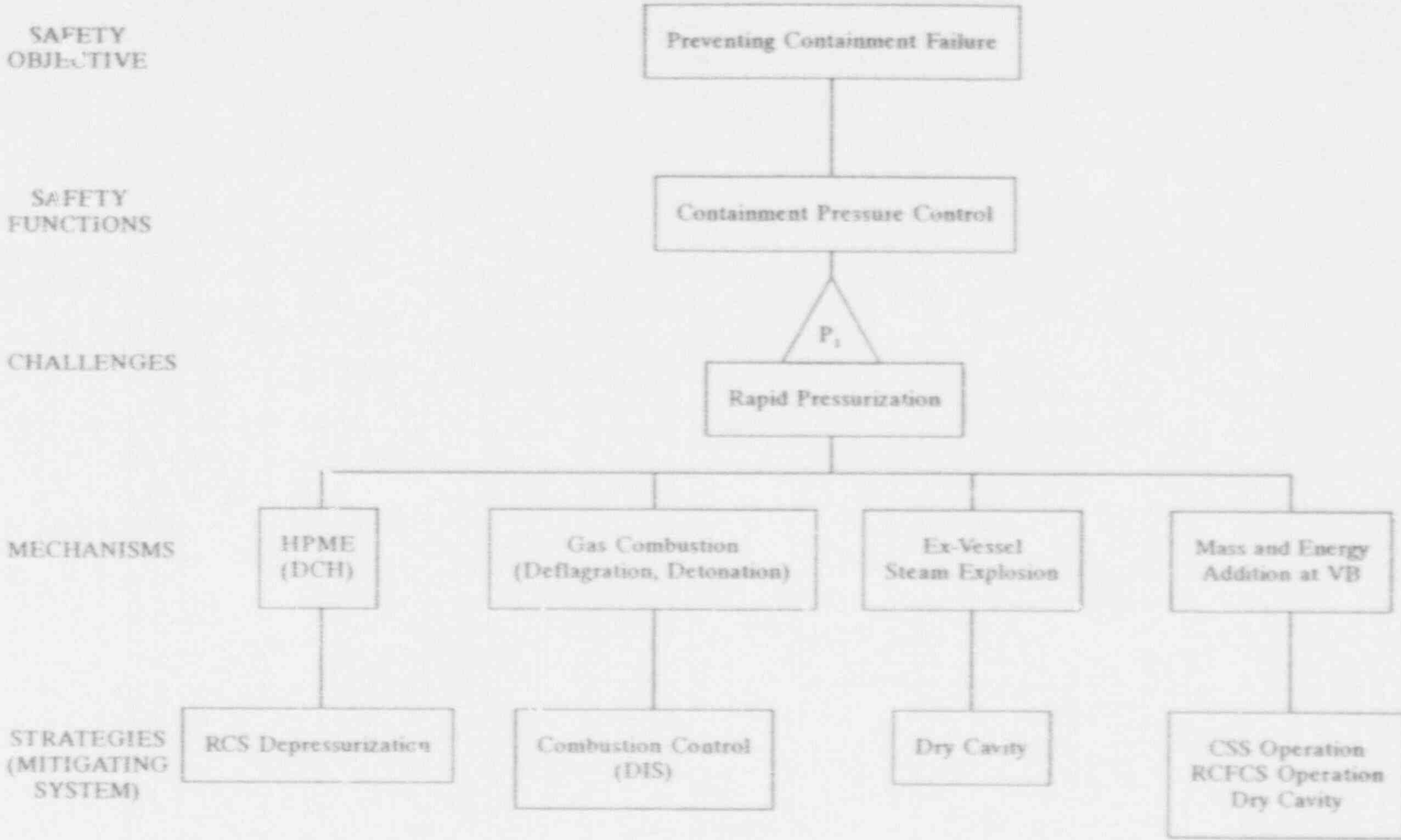


Figure 3.1 Safety Objective Tree for PWR Plants With Dry Containment Design (Continued)

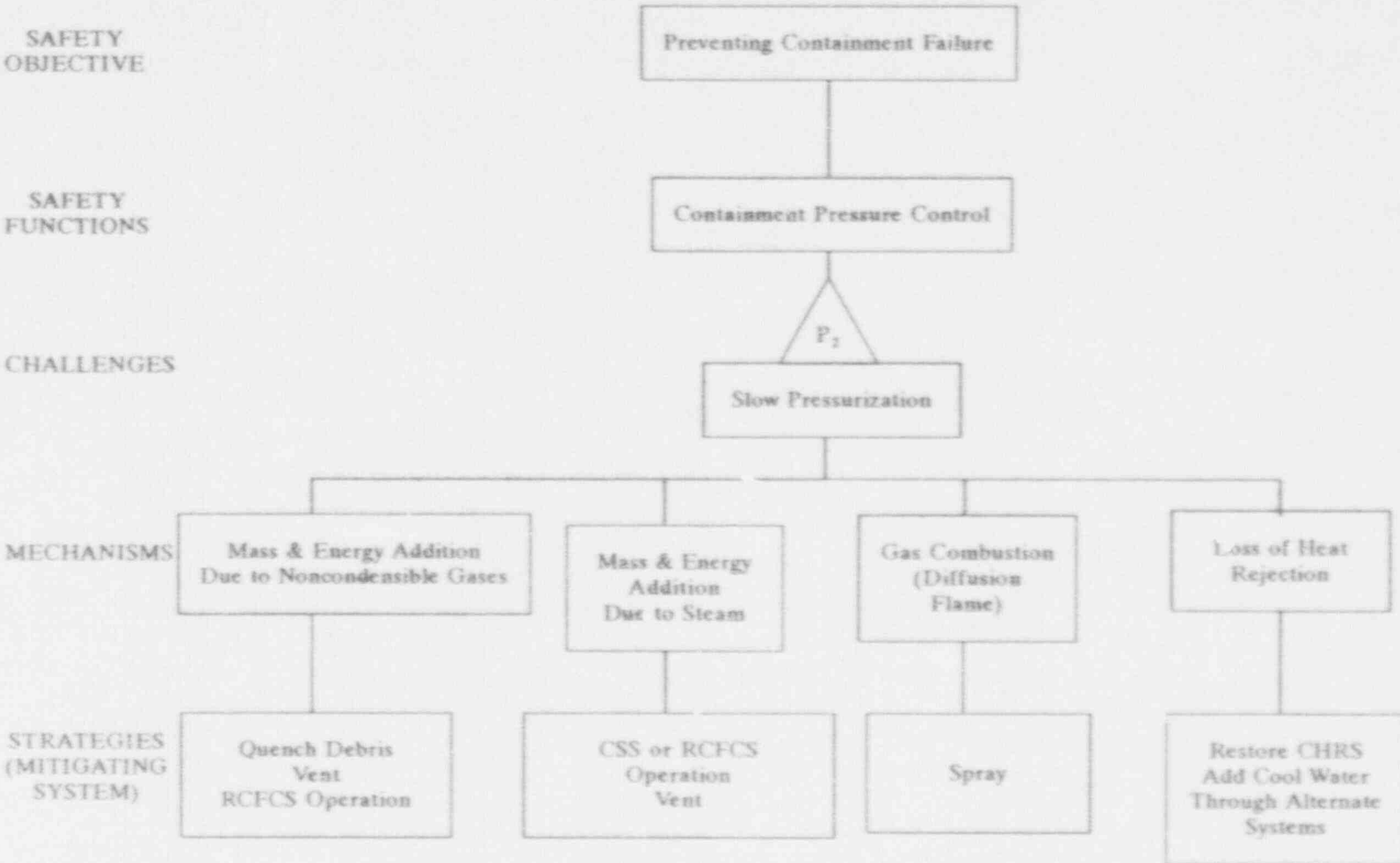


Figure 3.1 Safety Objective Tree for PWR Plants With Dry Containment Design (Continued)

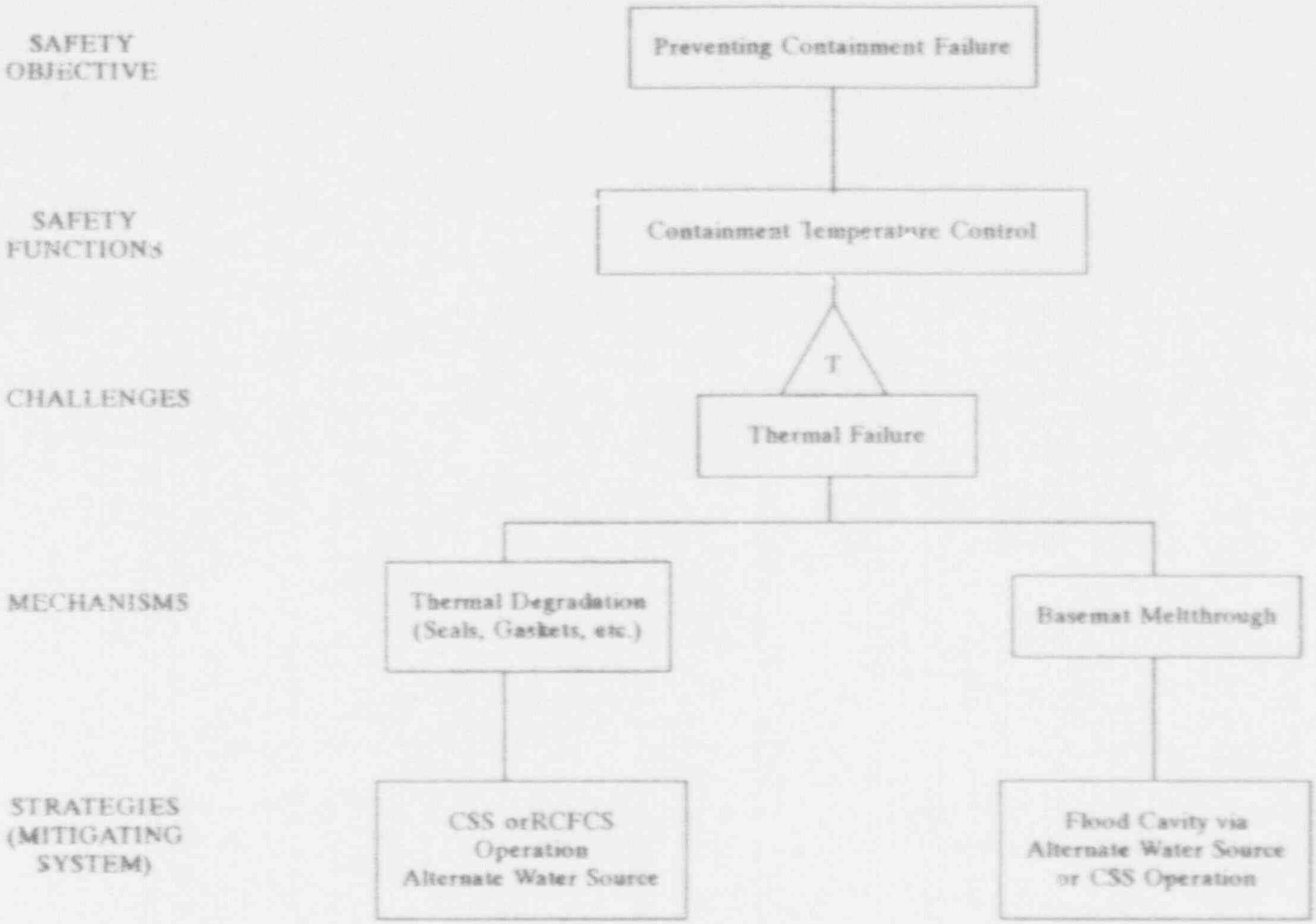


Figure 3.1 Safety Objective Tree for PWR Plants With Dry Containment Design (Continued)

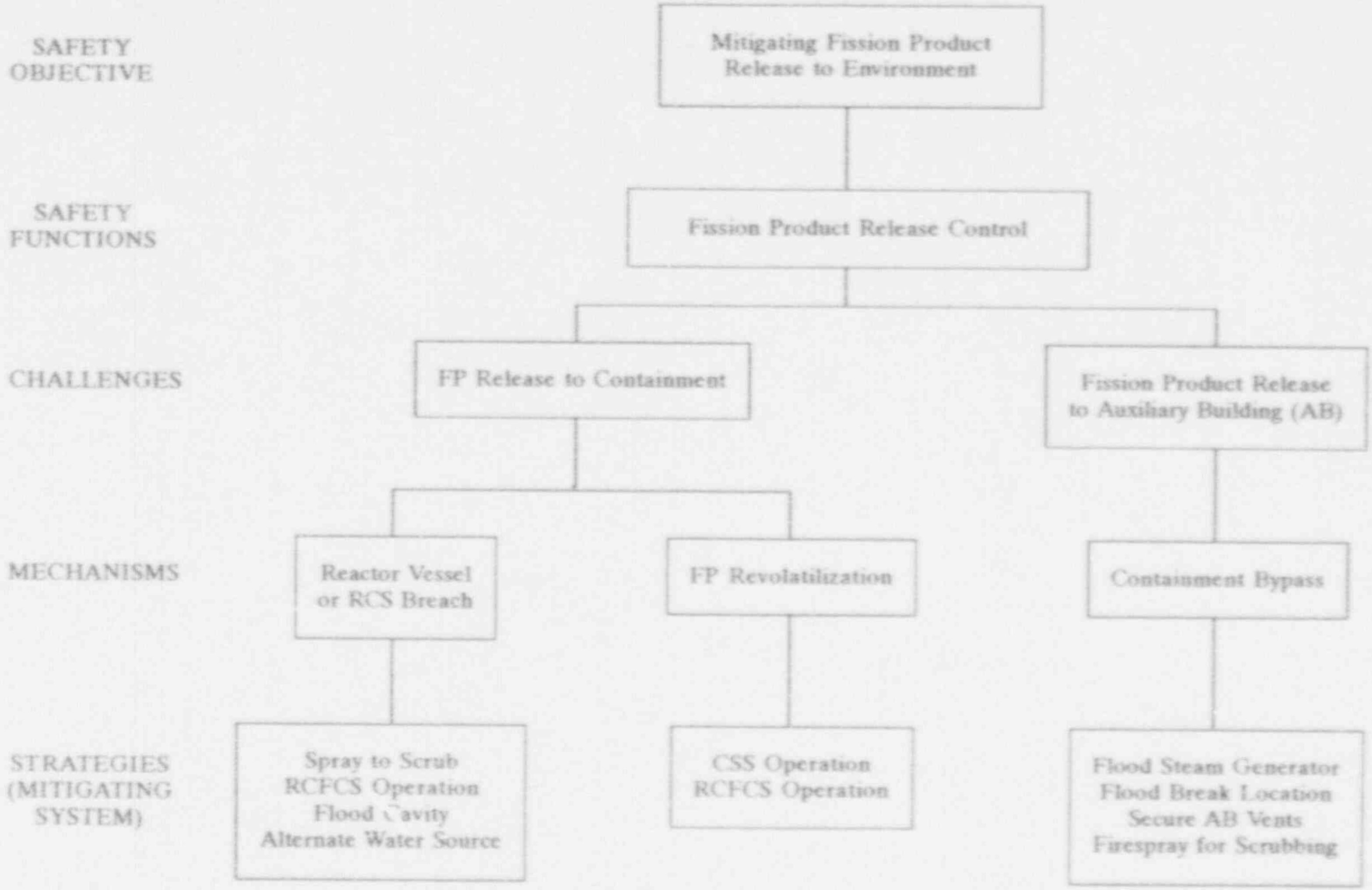


Figure 3.1 Safety Objective Tree for PWR Plants With Dry Containment Design (Continued)

An interfacing system LOCA (ISLOCA) refers to a class of accidents in which the reactor coolant system pressure boundary interfacing with a supporting system of lower design pressure is breached. If this occurs, the low pressure system will be overpressurized and could rupture outside the containment. This failure would establish a flow path directly to the environment or, sometimes, to another building of small pressure capacity. As the RCS loses coolant inventory, make-up will be provided from the refueling water storage tank (RWST). When the water in the RWST is depleted, recirculation cooling is not possible because the water has been lost outside the containment. Core damage will then occur if an alternative water supply is not provided. Depending on the accident sequences, the emergency core cooling system and other safety system may also fail, resulting in a rapid core melt with containment bypass.

BNL performed a detailed study of an interfacing systems LOCA for pressurized water reactors [8]. Three plants were selected: Indian Point 3 (Westinghouse), Oconee 3 (Babcock & Wilcox), and Calvert Cliffs 1 (Combustion Engineering). The interfacing lines which have been identified as potential ISL pathways include lines of low pressure injection, high pressure injection, residual heat removal suction, letdown and the core flooding tank (accumulator) outlet lines. The results of the BNL study indicate that the contributors from two groups of pipe lines, namely the Residual Heat Removal suction and Low Pressure injection lines, dominate the core damage frequency (CDF) due to ISLOCAs. The total contribution of ISLOCA events to CDF is generally less than a few percent of the overall CDF. However, they can potentially be important contributors to risk if core damage occurs because ISLOCAs may bypass the containment and allow radioactive material release directly to the environment.

For some plants, ISLOCAs are deemed to be not important because of the plant-specific configuration of the high-to-low pressure system interfaces, the location of the RHR pumps, and the RHR relief valves. If these are inside containment, many potential ISLOCAs will not bypass containment. For the Surry plant, three check valve failure scenarios were identified as the initiating event for this sequence. The frequency of the ISLOCA initiating event per reactor year is estimated as 3.8×10^{-7} for the Surry plant [2].

The basic procedures to control an ISLOCA considered for the accident management strategies involve early detection, isolation, or mitigation of the effects [9] and will be discussed in Section 3.3.4.

Steam generator tube rupture induced by high temperatures represents another containment bypass event. Analyses have indicated a potential for very high gas temperatures in the reactor coolant system during accidents involving core damage with the primary system at high pressure. For example, a RELAP5 analysis for the Bellefonte plant (a PWR designed by Babcock & Wilcox) showed that natural circulation flow of the gases through the primary loops could cause a large temperature increase in the reactor coolant piping system [10]. The high temperature could fail the steam generator tubes long before the core begins to relocate. As a result of the tube rupture, the secondary side may be exposed to full RCS pressures. These pressures are likely to cause relief valves to lift on the secondary side. If these valves fail to reclose, an open pathway from the vessel to the environment can result.

3.2.2 Direct Containment Heating (DCH)

In certain core damage accidents, such as a small LOCA or station blackout, the core debris could penetrate the reactor vessel when the primary system is at high pressure. Under these conditions, the molten core materials ejected under high pressure are likely to be dispersed out of the reactor cavity into the containment volume as small particles and they can quickly transfer thermal energy to the containment atmosphere. During the process, the metal contents of the ejected material, mostly zirconium and steel, can react with oxygen and steam to generate chemical energy and hydrogen. Hydrogen combustion could occur if the conditions are favorable. This direct energy exchange via (a) melt-atmosphere heat transfer, (b) melt-steam chemical reaction, (c) melt-oxygen chemical reaction, and (d) hydrogen combustion can lead to rapid containment pressurization which has the potential to fail some containments under certain accident conditions.

A review of DCH phenomenology and its impact on containment loading has been made by Ginsberg and Tutu [11]. The review shows that there is a large uncertainty in predicting DCH loads, which is caused by uncertainties in initial conditions, reactor cavity thermal-hydraulic behavior, transport of the melt in subcompartments, and hydrogen combustion during the process.

In support of the NUREG-1150 project, BNL and Sandia performed DCH parametric studies for the Zion and Surry plants, respectively. The CONTAIN-DCH 1.10 version code [12] which has modifications which parametrically characterize DCH phenomena was used in both studies. In BNL's Zion analysis [13], two accident sequences, TMLB' (station blackout) and S_2D (small break LOCA with loss of all coolant injection) were considered. The two accidents, representing high-pressure and low-pressure scenarios, served to define the direct containment heating initial conditions. A wide range of initial conditions and phenomenological assumptions were selected to represent the current uncertainties in DCH. The parameters varied in the sensitivity study included: primary system pressure at vessel failure, core melt inventory, melt and steam flow rates through the reactor cavity, melt droplet size, melt trapping rate, extent of hydrogen combustion, quenching of trapped debris, and co-dispersal of water from the reactor cavity.

In Sandia's Surry analysis [14], two station blackout accident sequences, representing high-pressure and low-pressure scenarios, were considered. The first case (referred to as FP) was an early station blackout with the auxiliary feedwater assumed to be unavailable and the primary system assumed to remain fully pressurized at about 16 Mpa (2320 psia) until vessel breach. The second case (referred as PS-LOCA) was a long-term station blackout in which auxiliary feedwater was assumed to be available and the secondary-side was depressurized. A pump seal LOCA occurred and the primary system was partially depressurized to about 5.1 Mpa (740 psia). The sensitivity studies identified that the containment loading was affected by the reactor vessel initial pressure, melt trapping in subcompartments, hydrogen burn under high temperatures, the quantity of steam available for thermal interaction and the melt blowdown rate, etc. The results are similar to that reported in the Zion study.

Although the parametric studies performed for Zion and Surry indicate a large uncertainty in containment loading due to DCH, it can generally be concluded that containment integrity would be threatened if a large fraction of melt is involved in the heating process and a hydrogen burn occurs. However, containment loading is plant specific, as there are a number of specific parameters likely to be significant for DCH, including containment size, strength, and internal layout.

It should be noted that there are two important issues which add to the uncertainty associated with DCH events: hydrogen combustion and cavity flooding.

The high temperatures induced by the DCH processes contribute to a large uncertainty regarding hydrogen combustion. There is some experimental and theoretical evidence indicating that at high temperatures, the requirements on atmospheric composition for flammability or detonability are much less stringent than at low temperatures [15]. Thus, it is likely that hydrogen combustion would take place during the DCH event, particularly in the containment dome region which is rich in oxygen. The importance of a hydrogen burn is illustrated in Figure 3.2, in which the DCH pressure rise was computed by a thermodynamic adiabatic equilibrium model developed by Ginsberg and Tutu [11]. The model provides an upper limit to the DCH containment pressure loading and serves as a basis for comparison. In the analysis it was assumed that all melt which exists in the reactor vessel is involved in DCH (i.e., no trapping). It is seen that if all the available hydrogen was to burn, then the computed upper limit containment pressure rises are containment-threatening over a broad range of participating melt mass. The computed containment pressure rises are considerably lower if no hydrogen burn is assumed. For Zion, the predicted DCH pressure is lower than the estimated mean failure pressure if hydrogen burns are assumed not to occur. For Surry, the containment-threatening loads are predicted only for a high fraction of melt participation if hydrogen burns are assumed not to occur.

The effects of the presence of water in the reactor cavity on containment pressurization during a DCH event also involve a large uncertainty. This uncertainty has been addressed in the NUREG-1150 report [2]. It states that at least two scenarios are conceivable when water interrupts the pathway for debris dispersal following reactor vessel breach. One scenario is that one or more steam explosions will occur after only a fraction of the debris has been injected into the cavity and that the cavity water will then be dispersed ahead of the bulk of the injected debris.

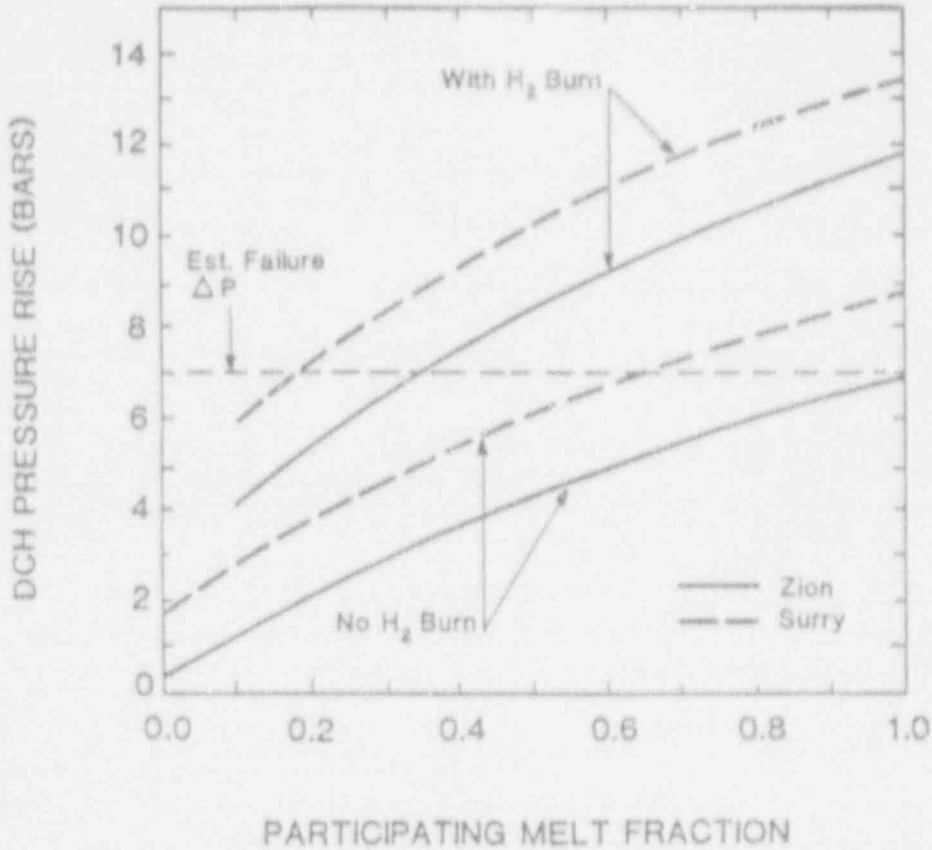


Figure 3.2 Effect of H_2 Burn on DCH Pressure Rise Based on Thermodynamic Adiabatic Equilibrium Model (Reference 11).

injected into the cavity and that the cavity water will then be dispersed ahead of the bulk of the injected debris. Another scenario is that water will be co-dispersed with the debris, existing in the cavity region as small droplets intermixed with the transported debris, steam and hydrogen. The water may quench the debris, mitigating the effects of direct containment heating. On the other hand, the steam generated by the co-dispersed water may enhance the containment pressurization rate, or may react with unquenched debris to generate hydrogen and liberate chemical energy. The CONTAIN parametric studies [13,14] show that the containment pressure increases if water is present in the reactor cavity. The analyses imply that the quenching of trapped debris releases additional steam into the containment atmosphere which directly contributes to the incremental pressure rise. Furthermore, the addition of this steam into the lower regions of containment causes additional trapped hydrogen in these regions to move upward into the oxygen rich upper dome where it can burn. A series of limited tests referred to in Reference 40 also recorded higher peak pressures with water in the cavity.

The CONTAIN analyses indicate a trend of a lower DCH pressure rise with decreasing initial pressure in the reactor coolant system (RCS). An example is given in Figure 3.3 for the Zion plant [21]. The CONTAIN result is driven, in part, by the assumed increase of melt droplet diameter with decreasing RCS pressure and the consequent reduction of surface area available for heat and mass transfer. The trend, however, is also believed to be the result of the influence of RCS steam inventory and the effect of the resulting steam flows on the processes of hydrogen production and convection. The pressure rise is also related to the blowdown rate. A slower rate would allow more energy to be transferred to heat sinks. Thus, it appears that RCS depressurization could be considered as a preventive strategy in combatting DCH.

3.2.3 Combustion

During a severe accident in an LWR, oxidation of the metallic components of the reactor core will produce hydrogen. Hydrogen combustion in the containment building could produce pressure and temperature levels that may threaten the integrity of the containment boundary. The threat to containment depends on the details of the accident sequence and the containment design. Because the typical PWR dry containment has a higher design pressure and larger containment volume in comparison with other designs, it has a greater ability to accommodate the large quantities of hydrogen associated with a severe accident than BWR containments and PWR ice-condenser containments. Thus, the PWR dry containments were excluded from the NRC interim hydrogen rule which requires the control of hydrogen produced by the metal/water reaction (MWR) of an equivalent of 75% of the cladding

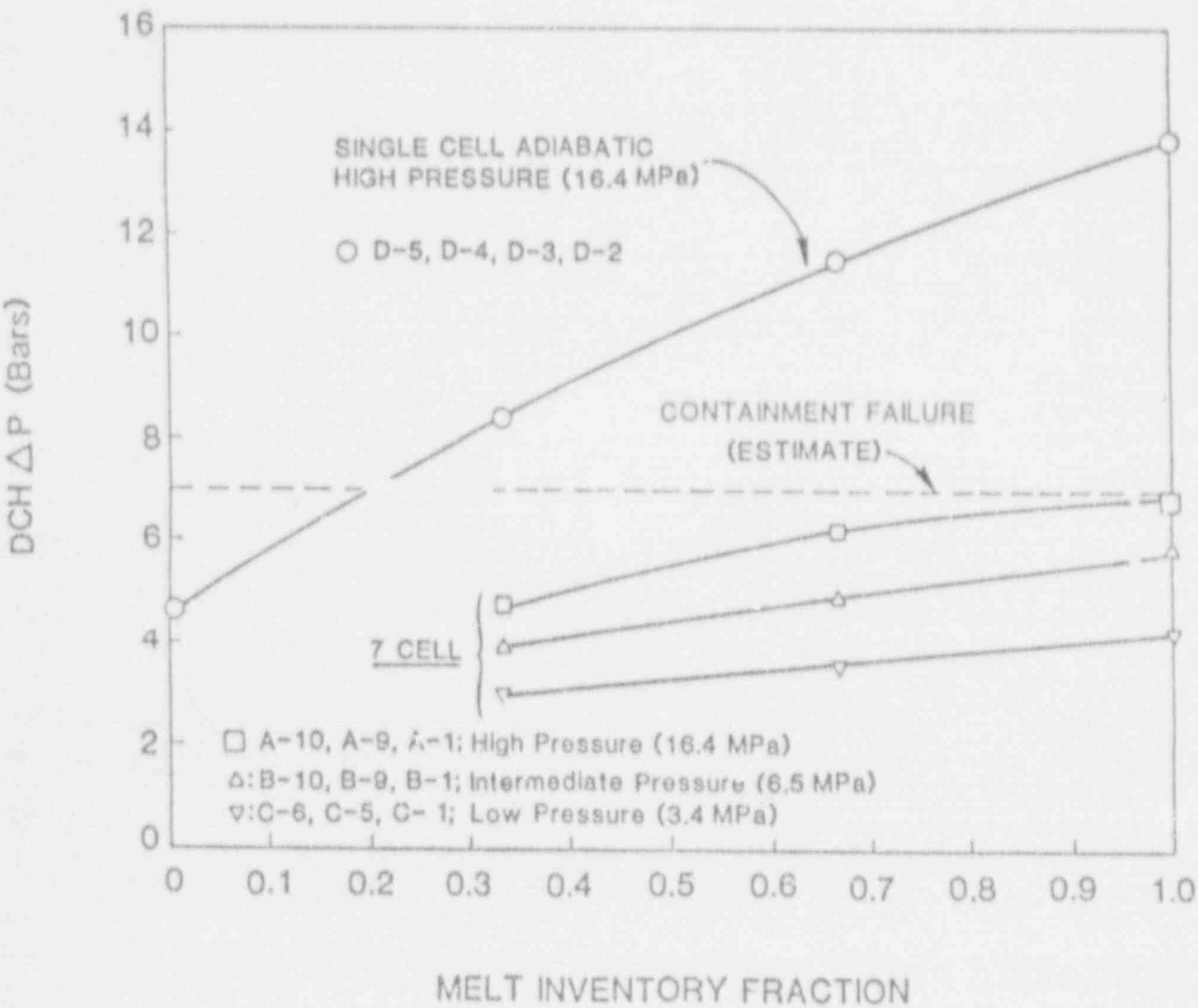


Figure 3.3 Zion DCH Calculation Results for Various RCS Pressures

surrounding the active fuel. In the NUREG-1150 study, hydrogen combustion is not identified as a dominant contributor to early containment failure for either the Zion or the Surry plant. While hydrogen combustion alone is not considered to be a severe challenge for PWR dry containments, the combined effects of hydrogen combustion with DCH and steam spike could threaten the containment integrity under certain conditions.

Combustion phenomena can involve hydrogen burns prior to reactor vessel breach, and combined hydrogen/carbon monoxide burns after the reactor vessel breach and the initiation of corium-concrete interaction. These phenomena involve many issues, such as in-vessel and ex-vessel hydrogen production, concrete erosion and CO production, ignition probability, detonation probability, peak pressure rise from a deflagration, dynamic load from a detonation, and thermal impact on safety related equipment. These issues have been discussed in Reference 16 for PWR dry containments.

Potential hydrogen generation equivalent to 75% and 100% fuel-cladding oxidation are given in Table 3.1 for a group of PWR plants with atmospheric and subatmospheric containment design. The quantities of hydrogen that would be

Table 3.1 Hydrogen Concentrations in PWR Dry Containment (Reference 16)

Plant	Thermal Power, MW	Free Volume (10^6 ft^3)	Clad Mass (lbm)	75% Oxidation: H_2 Mass (lbm)	75% Oxidation: Vol.%	100% Oxidation: H_2 Mass (lbm)	100% Oxidation: Vol.%
Ginna	1300	0.97	22440	738	13	984	17
Kewaunee	1650	1.32	24443	804	11	1072	14
Prairie Island	1650	1.32	24443	804	11	1072	14
Turkey Point	2200	1.55	36300	1194	13	1592	17
Summer	2775	1.84	38280	1257	12	1676	15
Maine Yankee	2640	1.86	53768	1768	16	2358	20
Oconee 1,2,3	2568	1.9	42200	1388	13	1850	16
Robinson	2300	1.95	36300	1194	11	1592	14
Rancho Seco	2772	1.98	42200	1388	13	1850	16
Crystal River 3	2452	2.0	42200	1388	13	1850	16
Farley	2652	2.0	38230	1257	11	1676	14
Comanche Peak	3411	2.5	50913	1614	12	2233	15
Indian Pt. 2	2758	2.61	44600	1467	10	1956	13
Zion	3250	2.60	44550	1466	10	1954	13
Indian Pt. 3	3025	2.61	41993	1382	10	1842	12
Diablo Canyon 1,2	3411	2.63	46993	1545	10	2061	13
WNP-1	3800	3.09	51450	1692	10	2256	13
Bellefonte	3620	3.35	51450	1692	9	2256	12
South Texas	3817	3.40	54840	1804	10	2405	12
Surry	2650	1.80	36300	1194	17	1592	21
Millstone	3425	2.30	45296	1490	16	1986	20

Note:
1. Free volume and clad mass are from FSAR of each plant.
2. Vol.% is computed for dry atmosphere at 100F

generated from a 75% reaction vary from about 360 Kg (800 pounds) for smaller plants, such as Ginna, Kewaunee and Prairie Island, to about 770 Kg (1,700 pounds) for larger plants, such as South Texas, WNP-1 and Bellefonte. The quantity of hydrogen from 100% clad oxidation ranges from about 454 Kg (1,000 pounds) for smaller plants, to more than 900 Kg (2,000 pounds) for the larger plants. Additional hydrogen could be generated from the oxidation of steel in the reactor cavity region.

Estimates of containment peak pressures due to hydrogen deflagration are shown in Table 3.2 [16] for a number of plants. The table also includes the containment design pressure and estimated failure pressure. Since the containment failure pressure has not been analyzed for many of the plants, it is taken as 2.5 times the design pressure. Table 3.2 indicates that the estimated peak pressure due to a global hydrogen burn based on a 75% MWR is within the estimated containment capacity. Therefore, it seems unlikely that containment integrity would be threatened by a hydrogen deflagration from a 75% MWR in these containments examined. Table 3.2 also indicates that a hydrogen burn resulting from a 100% MWR would produce pressures very close to the estimated containment capacity. The above adiabatic pressure increase due to a global hydrogen deflagration represents the upper limit of potential containment loading. Under non-adiabatic conditions, the pressure increase is expected to be smaller.

Table 3.2 Estimated Adiabatic Pressure Rise Due to Hydrogen Deflagration

Plant	Design Pressure	Estimated Failure Pressure	Pressure Rise (a)	Pressure Rise (b)
Ginna	60	150	108	133
Kewaunee	46	115	93	113
Prairie Island	41	103	93	113
Turkey Point	59	148	109	135
Maine Yankee	55	138	127	159
Oconee 1,2,3	59	148	105	129
Summer	57	143	101	123
Robinson	42	105	93	114
Rancho Seco	59	148	102	125
Crystal River 3	55	138	101	126
Farley	54	135	95	116
Comanche Peak	50	125	99	121
Indian Pt. 2	47	118	88	107
Zion	47	118	88	107
Indian Pt. 3	47	118	85	103
Diablo Canyon 1,2	47	118	91	110
WNP-1	52	130	87	105
Bellefonte	50	125	83	99
South Texas	57	143	85	103
Surry	45	113	84	106
Millstone	45	113	82	104

Note:
1. Pressures in psig.
2. Design pressure, clad mass, and free volume are from FSAR of each plant.
3. Failure pressure = 2.5 x design pressure.
4. Pressure rise is based on (a) 75% clad oxidation, (b) 100% clad oxidation.

Although PWR dry containment designs are generally not highly compartmentalized, their large free volume means that a relatively long time is needed to achieve a well mixed atmosphere. The average hydrogen volume concentrations based on the assumption that hydrogen is well mixed with dry air at 100 F (311 K) are included in Table 3.1. The assumption of mixing with dry air tends to maximize the potential hydrogen concentration and is unlikely to occur during an actual accident. Under accident conditions, it is likely that steam will be released simultaneously from the reactor vessel to the containment. The hydrogen concentration corresponding to a dry atmosphere could be reached only if all the steam released to the containment is condensed by the containment heat removal systems. Thus, caution must be exercised in activating the containment spray system or fan cooling system late during the accident when a large quantity of combustible gases has accumulated in the containment. To the best of their ability, the plant staff should calculate prevailing hydrogen and steam concentrations in the containment based on containment temperature and pressure, as well as estimates of clad damage and water inventory in the containment. For a given spray duration and flow rate, another estimate can be made for hydrogen-steam concentrations subsequent to spray activation. Such estimates can be helpful for making a decision regarding spray activation.

Table 3.1 reveals that at 100% Zircaloy oxidation the global average hydrogen concentration in the containment is higher than the lean detonation limit as indicated by the National Research Council Report [17]. The subatmospheric plants, which operate at about 10 psia and have less air to dilute the hydrogen concentration, appear to be potentially more vulnerable to combustion.

One of the important issues of combustion in PWR dry containment is the non-uniform mixing of gases which could lead to local detonation. Multiple compartment analyses using the CONTAIN code show that it could take hours to achieve uniform hydrogen distribution by the intercell flow mixing due to natural circulation processes [16]. Prior to achieving a uniform distribution, there are situations where high concentrations of combustible gases could exist in subcompartments. An example is given in Figure 3.4 which illustrates the transient mole fractions of gases in the cavity region for a small break LOCA sequence [18] in Zion. It is seen that the atmosphere contains a large fraction of hydrogen and carbon monoxide during a brief period from about 5000 seconds to 6000 seconds. A peak hydrogen fraction occurs at about 5271 seconds, and a peak carbon-monoxide fraction at about 5811 seconds. The corresponding fractions of gases are shown below:

Time, s	5271	5811
H_2%	14	7.5
CO%	0	22
O_2%	5.5	4.5
Steam%	60	50

During this period of about 1000 seconds, the combined hydrogen and carbon monoxide fractions are between 0.14 and 0.30. The hydrogen may be above the detonation limit reported in literature. CO may also burn. It is not clear whether the steam fraction is high enough to make the atmosphere inert and there may be sufficient oxygen to initiate combustion. CONTAIN analysis also indicated that the atmosphere temperature in the cavity region (the source compartment) is between 750 K to 1340 K. The high temperature and high fractions of combustible gases would lead to a favorable condition for local detonation. A local detonation could threaten the containment integrity.

It should be pointed out that a PWR dry containment with a steel shell design is particularly vulnerable to missiles generated due to a local detonation in the containment [19,20]. If the internally generated missiles hit the steel shell, local mechanical damage may occur. If the mechanical damage coincides with a global load caused by internal pressurization, a failure of the containment is possible. At present, the U.S. has two PWR plants with spherical steel shell designs, and seven plants with cylindrical steel shell designs.

Activation of the containment fan cooling system can mitigate the potential for local detonation by promoting flow mixing in the containment. The degree of mixing, however, depends on the fan capacity and containment volume. Significant mixing will occur in the time required to turn over the entire containment volume. The times required to process one containment volume by fans are shown in Table 3.3 for several PWR plants with large dry containments. (Note that subatmospheric plants do not have recirculation fans as part of their engineered safety features.) Table 3.3 indicates that good mixing by fans alone could take between 10 to 30 minutes for the PWR containments examined. Containment sprays, which promote turbulent motion in the atmosphere, can also enhance the mixing process. However, both fan coolers and sprays tend to de-inert the atmosphere by condensing steam and thus increase the chances for combustion.

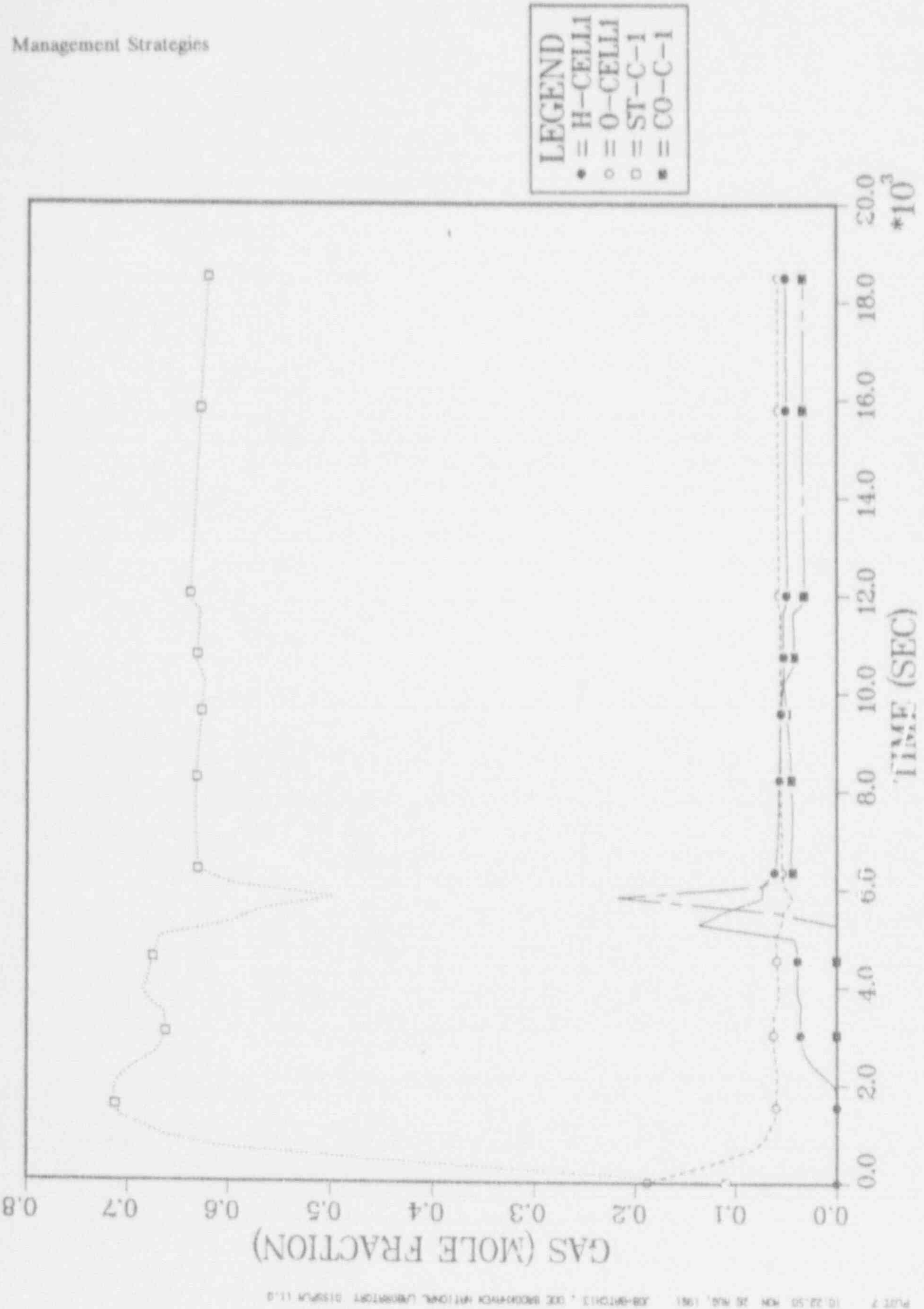


Figure 3.4 Gas Mole Fraction in Reactor Cavity Region for a Small Break LOCA Sequence With Dry Cavity Configuration

Table 3.3 Comparisons of Containment Fan Capacity and Turnover Time

Plant	Number of Fans	Fan Capacity (cfm) (Each)	Free Volume (10^6 ft^3)	Number of Fans Operating	Turnover Time (min)
Robert Ginna	4	17,000	0.972	3	19
Turkey Point	3	25,000	1.55	2	30
Palisades	4	60,000	1.60	3	9
Nuclear One-2	4	50,000	1.78	3	12
Virgil Summer	4	61,500	1.84	3	10
Robinson 2	4	65,000	1.95	3	10
Rancho Seco	4	40,000	1.98	3	17
Crystal River 3	3	54,000	2.0	2	19
Calvert Cliffs	4	55,000	2.0	3	12
Farley	4	40,000	2.0	3	17
Haddam Neck	3	50,000	2.23	2	22
San Onofre-2	4	31,000	2.36	3	25
Indian Pt. 2	5	65,000	2.61	4	10
Diablo Canyon	5	47,000	2.63	4	14
Waterford-3	4	35,000	2.66	3	26
Davis-Besse	3	58,000	2.87	2	16
South Texas 1,2	6	50,825	3.4	5	13
Zion	5	53,000	2.6	3	16
TMI-2	5	47,000	2.0	4	11

Note:
1. Fan capacity and free volume are obtained from the FSAR of each plant.
2. For some plants, the fan capacities given in FSAR are based on the normal operation condition. The capacity will be reduced under severe accident conditions.

3.2.4 Steam Explosion

The term "steam explosion" refers to a phenomenon in which molten fuel rapidly fragments and transfers its energy to the coolant resulting in steam generation, shock waves, and possible mechanical damage. To result in a significant safety concern the interaction must be very rapid and must involve a large fraction of the core mass. If such events were to take place within the reactor pressure vessel, missiles could be generated which might penetrate the containment and allow early release of radioactive material. In the Reactor Safety Study (WASH-1400) this mode of containment failure due to in-vessel steam explosion was denoted as the alpha-mode failure.

In-vessel steam explosions and direct containment heating are the two major physical phenomena contributing to the low estimate of early containment failure of large dry containments. In-vessel steam explosions, which are controlled by the triggering mechanism, are more likely to occur when the primary system is at low pressure. On the other hand, direct containment heating is associated with high pressure sequences. The estimated relative contributions from these two phenomena for each plant damage state (PDS) of the Zion plant are given in Table 3.4 [21]. Any strategy for mitigating the DCH event by RCS depressurization would increase the probability of in-vessel steam explosion.

The possibility of a steam explosion is not confined to the in-vessel phase of a severe accident. When water is present in the reactor cavity at the time of vessel failure, contact of molten core debris with water may result in a steam explosion. However, in the NUREG-1150 report [2], the Zion and Surry containments were not assessed to have significant vulnerability to impulse loads from ex-vessel steam explosions since accelerated water from the cavity would not directly contact structures that are both vulnerable and essential to the containment function.

Table 3.4 Comparison of the Contributions From α-Mode and DCH to Early Zion Containment Failure (Per Core Damage) (Reference 21)

PDS	α	DCH	Others	Total
SBO	5.910E-03	1.574E-02	3.183E-03	2.483E-02
(%)	(23.80)	(63.39)	(12.81)	(100.00)
Transients	3.505E-03	5.191E-03	3.058E-03	1.175E-02
(%)	(29.82)	(44.16)	(26.02)	(100.00)
LOCAs	7.983E-03	1.706E-03	4.166E-03	1.386E-02
(%)	(57.62)	(12.31)	(30.07)	(100.00)

3.2.5 Mass and Energy Addition at Vessel Breach

After the reactor vessel has been breached and the core debris discharged into containment, a large quantity of mass and energy is added to the containment. The masses are hydrogen, steam, water, and core debris. The energy sources are (1) internal energy of these masses, (2) chemical energy potentially available via oxidation of the core debris, and (3) decay heat associated with the core debris. The masses and internal energy of the hydrogen, steam, and water will not present any threat to a large PWR containment. The chemical energy via oxidation of the core debris is the DCH issue which has been discussed in Section 3.2.2. The internal energy and decay heat of the core debris, which can be transferred to the containment atmosphere by vaporization of available water in the cavity region or on the containment floor can impose a high pressure loading on the containment. This pressure loading is often referred to as a steam spike. The availability of water depends on the accident sequence and containment design. For example, the cavity would be relatively dry for sequences with early melt (failure of ECC injection) and no containment heat removal (failure of sprays).

The Containment Loads Working Group (CLWG), an ad hoc committee formed by the NRC, investigated the issue of a steam spike for the Zion containment [22]. The issue was defined as Standard Problem No. 1 (SP-1), in which the rapid quenching of the melt as it is being released into the reactor cavity was studied. The study showed that, on an equilibrium, adiabatic basis, the containment pressure due to a steam spike would be in the range of 48 to 96 psia. These results would be reduced if the effects of passive heat sinks are considered. Thus, it can be concluded that the steam pressure spike induced failure of a PWR large dry containment at the time of vessel failure is an event of relatively low probability.

3.2.6 Overpressurization Due to Noncondensible Gases and Steam

Without containment heat removal, the containment would fail by pressurization due to the addition of steam and noncondensible gases to the atmosphere. Even with containment heat removal pressurization will continue, albeit much more slowly, due to the noncondensibles. The noncondensible gases may evolve from corium-concrete interactions.

If the reactor vessel is depressurized, the core debris pouring out of the reactor vessel is likely to remain in the reactor cavity where it will interact with structural concrete. The erosion of concrete due to the thermal attack by core debris releases steam and CO_2 gas. (The quantities of steam and carbon dioxide released vary among different types of concrete, i.e., basaltic vs. limestone. For example, the basaltic aggregate concrete contains much less CO_2 and has a higher heat of fusion. It is expected that basaltic aggregate concrete would release less CO_2 gas and exhibit a relatively slower erosion rate.) As the gases flow upward through the molten debris, they may react with the metal components and generate H_2 and CO, which are combustible gases. Potential H_2 and CO burns could contribute to late pressurization failure.

Pressurization caused by corium-concrete interactions is governed by many factors, among which are the presence of water in the cavity and the potential for debris coolability. If the cavity is flooded with water prior to vessel failure, then, as the molten core materials fall into the water, rapid cooling and fragmentation is possible. This process could result in the formation of a coolable debris bed in which all the core decay heat is removed by boiling water and concrete attack is prevented. Under these circumstances, providing water flow to the cavity would replenish water loss due to boiling and ensure that the corium remains in a coolable and stable configuration.

If the cavity is initially dry and the core debris forms a deep bed, it could remain hot for a relatively long time and extensive concrete attack would occur. Under these circumstances, pouring water on top of the core debris may have the effect of rapidly cooling and stopping the concrete attack. However, experiments have shown that a crust can form on top of the molten core debris and effectively prevent the water from mixing with and cooling the debris. These experiments were performed at small scale and the stability of the crusts under the conditions of a severe accident in a power plant has not been established. Thus, the effect of pouring water on top of the core debris is uncertain [23]. Therefore, even though water flow is restored to the core debris, continued concrete attack with the generation of more combustible gases and release of radioactive material is still possible. The presence of water above the debris does have the advantage, however, of trapping a fraction of the radioactive material generated during the concrete attack, which would otherwise have reached the containment atmosphere.

Containment analyses indicate that late overpressurization failure can be expected to be a very slow process. The estimated containment failure time could extend anywhere from 15 hours to more than 30 hours after the initiation of the accident. Under these conditions, the loss of containment integrity could result via a "leakage-before-failure" mode rather than a catastrophic failure. The CPWG study [24] has pointed out that significant leakage may result due to high pressure and high temperature in the containment.

In the CPWG study [24], evaluations indicated that for the Zion plant the pressure-dependent leakage can delay or possibly prevent the containment from reaching its failure pressure for the TMLB' sequence with a flooded-cavity. For the Surry plant, the estimated containment pressures are not high enough to cause significant leakage. The pressure response is not significantly affected by leak areas less than 0.4 in^2.

3.2.7 Basemat Meltthrough

In the Zion PRA, it was assumed that for some sequences the reactor cavity would be flooded with water and hence a coolable debris bed would be formed, thereby preventing basemat meltthrough. The NUREG-1150 study [2], states that a significant likelihood exists for continuous debris-concrete interaction even if a replenishable water supply is available. Under these circumstances, the presence or absence of an overlying water pool is not expected to have much effect on the downward progression of the melt front.

The corium-concrete interaction rate depends on many factors, such as corium mass, composition and temperature, cavity configuration, and overlying water pool. The corium-concrete interactions reported in Reference 7 show that under certain initial conditions the axial concrete erosion rate for the Zion plant is estimated in the range of 0.035 cm/min to 0.087 cm/min depending on whether the cavity is flooded and when it is flooded. The basemat thickness for the Zion plant (a prestressed concrete containment) is 2.74 m (9 ft.). The complete erosion of the basemat would take several days. However, the basemat could lose its structural integrity before the complete erosion of its thickness. The basemat thickness for the Surry plant (a reinforced concrete containment) is about 3 m (10 ft.) and the estimated concrete axial erosion is in range of 0.068 cm/min. It also would take several days to penetrate the entire basemat.

3.2.8 Thermal Degradation

The design temperature for the Zion and Surry plants are 270 F and 150 F, respectively. These temperatures are much lower than the expected temperatures in containment during a severe accident. The primary impact of high temperature is the thermal degradation of containment penetration seals. The seals are made of elastomer compounds which could degrade when exposed to high temperatures for a long period of time. The failure of penetration seals could cause leakage to the containment. (This is referred to as the "leak-before-break" failure mode.) Data depicting seal life as a function of time at temperature for basic elastomer compounds used for containment construction is shown in Figure 3.5.

High temperatures in the containment are likely to occur late during the accident if (1) the core debris in the cavity assumes a noncoolable configuration and is not covered by water, and (2) the containment heat removal systems are not available. Under this situation, the corium-concrete interaction would release a large quantity of high-temperature gases and cause a severe thermal-environment in the containment for a long period. Figure 3.6 illustrates the CONTAIN predicted atmospheric temperature in 12 sub-compartments of the Zion plant for a small break LOCA sequence [18]. Beside the cavity region, temperatures in many compartments are above 500 K for several hours. Based on Figure 3.5, some seals may not be able to withstand these temperatures.

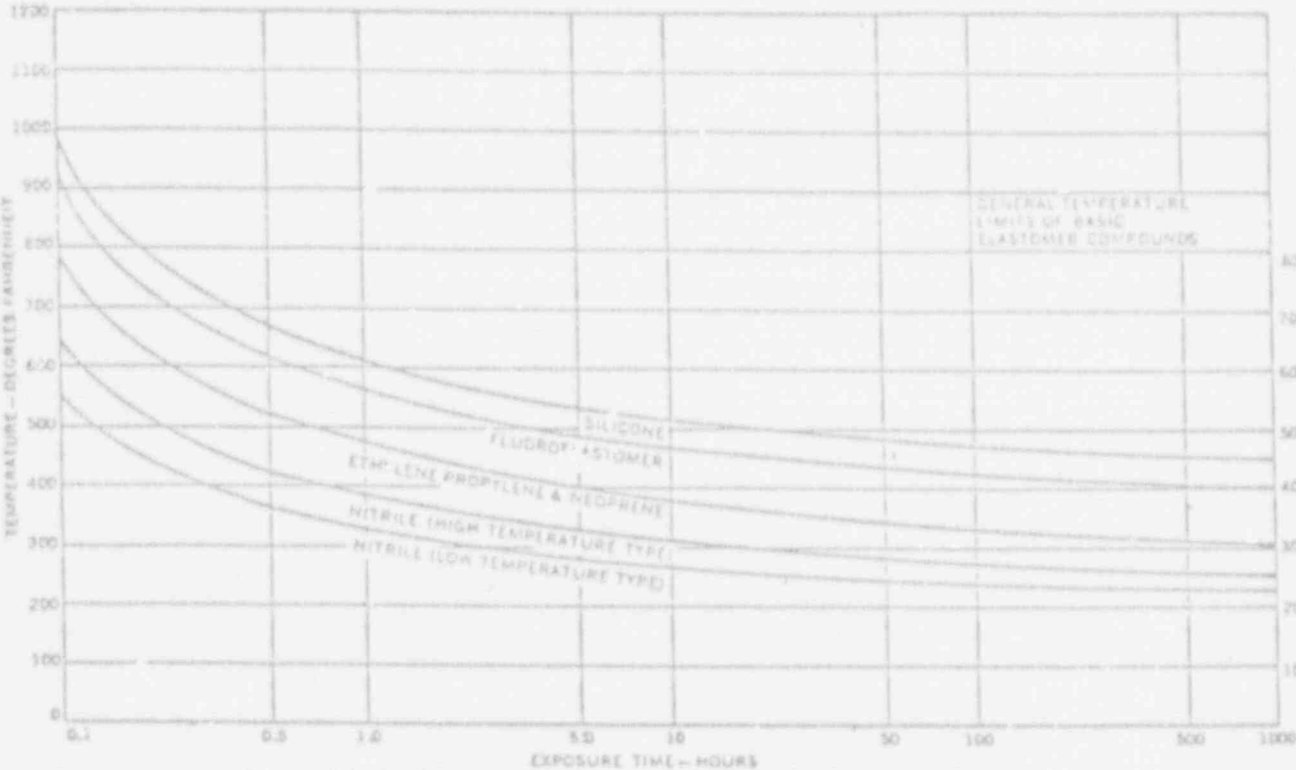


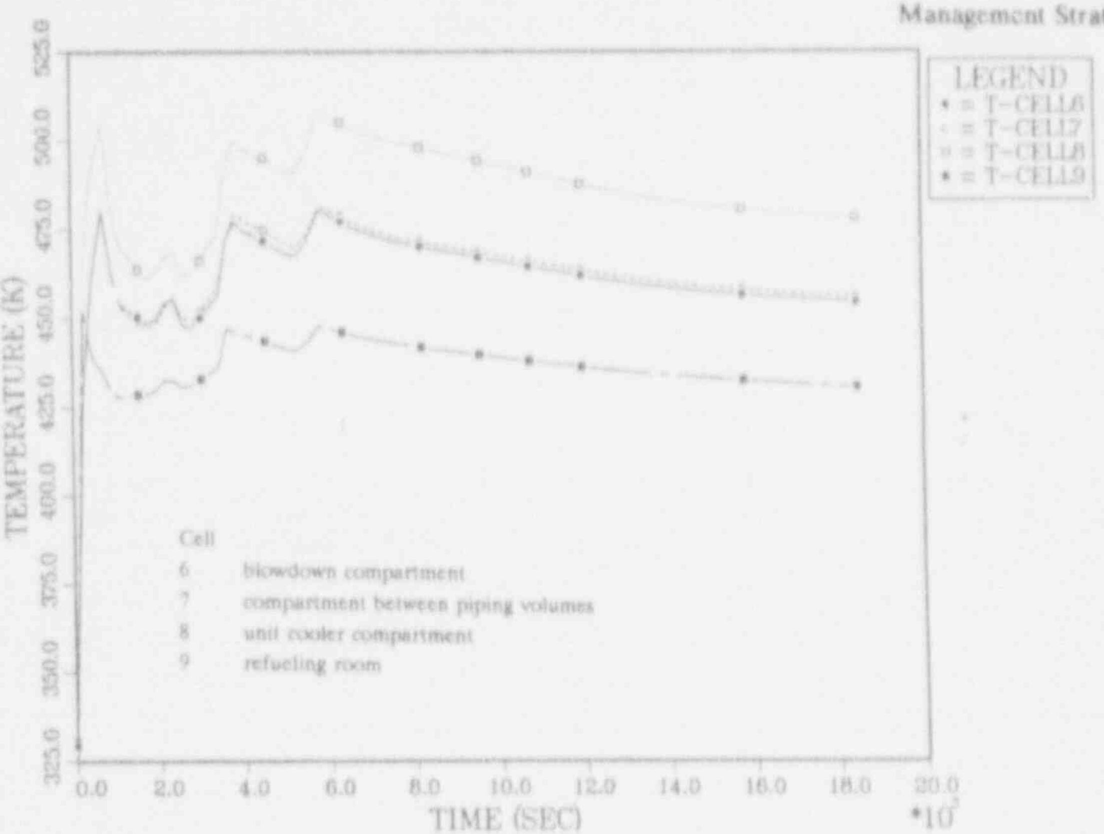
Figure 3.5 Seal Life as a Function of Time at Temperature (Reference 24)

Elimination or mitigation of the thermal degradation challenge could be achieved by the activation or restoration of the containment heat removal systems (sprays and/or fans).

3.3 Strategy Description

3.3.1 RCS Depressurization

The beneficial and adverse effects of RCS depressurization for core cooling have been extensively discussed [2] [37]. As far as containment integrity is concerned, depressurization of the RCS would reduce the threat of direct containment heating, and induced failures of steam generator tubes, and primary coolant piping, etc. However, RCS depressurization could increase the likelihood of steam explosion, although the frequency of containment failure caused by a steam explosion is predicted to be relatively low. Experiments reveal that a steam explosion becomes more likely at low ambient pressures [2].



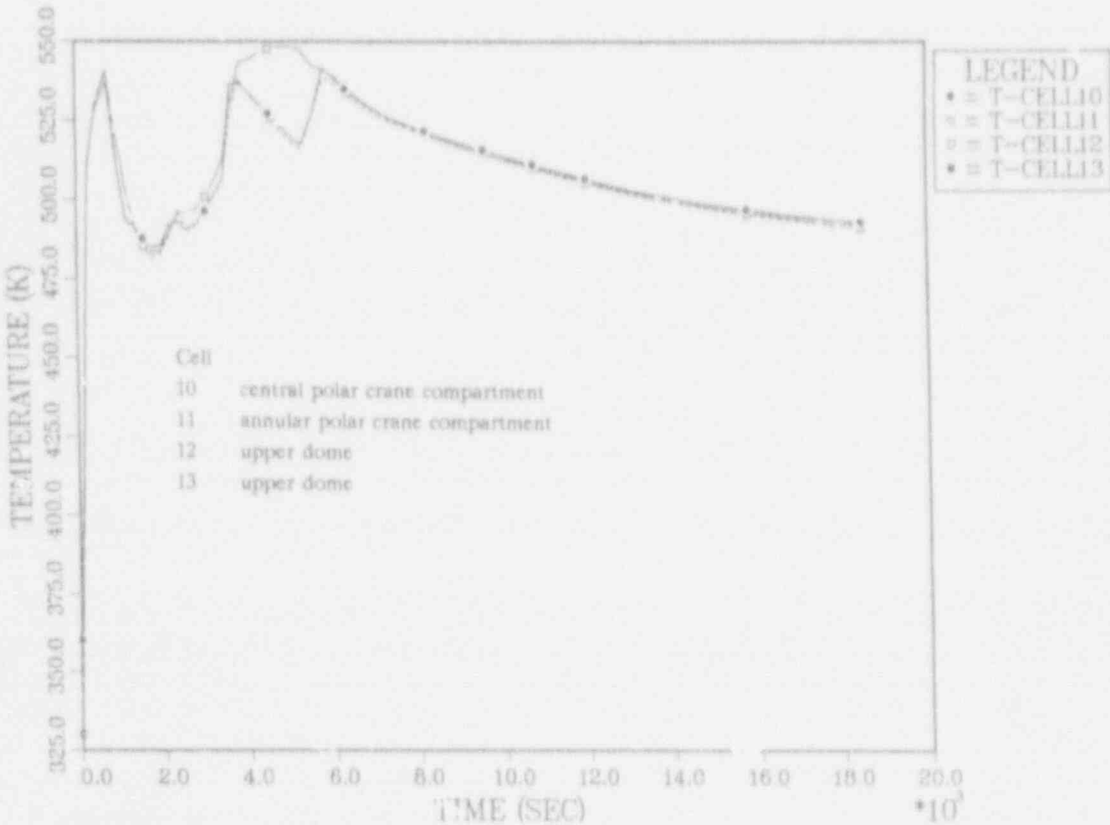


Figure 3.6 Containment Temperatures During a Small Break LOCA Sequence

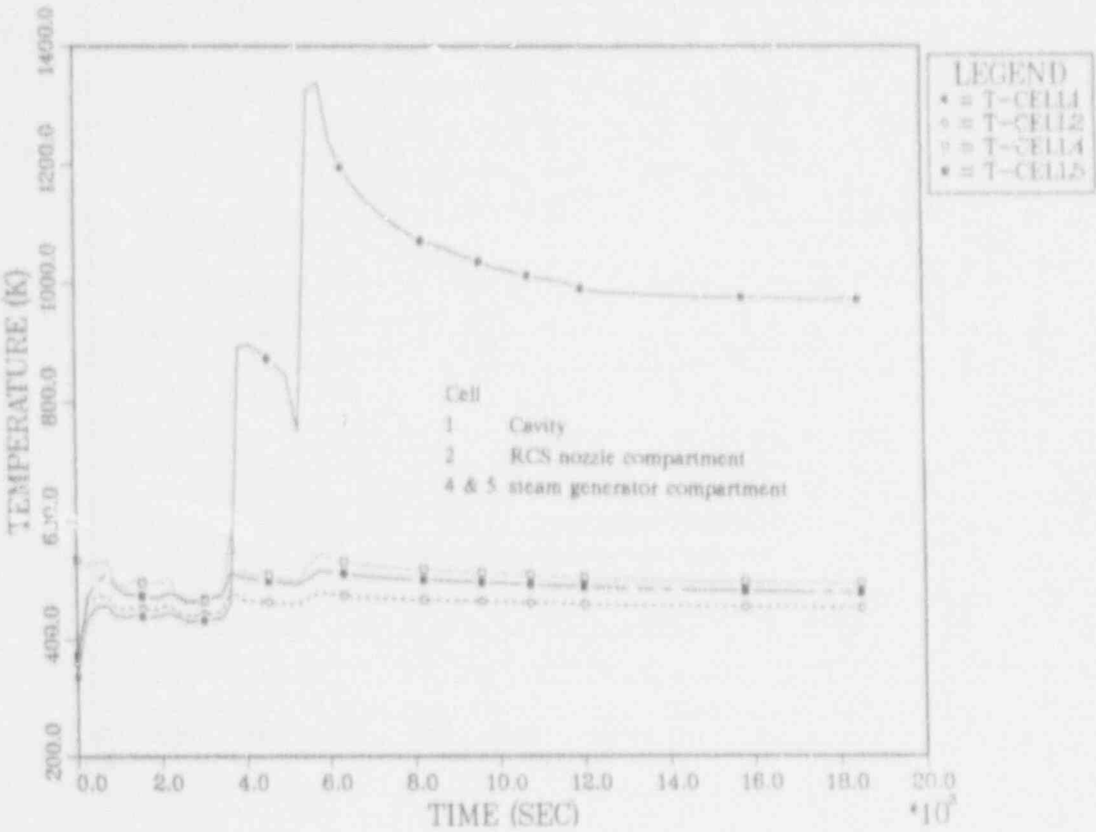


Figure 3.6 Containment Temperatures During a Small Break LOCA Sequence (Continued)

PWRs do not have a system specifically designed to manually depressurize the reactor vessel, however, the manual operation of all PORVs can effect limited depressurization. In a study of the feasibility of depressurization to avert DCH for the Surry plant, four operator actions that could depressurize the primary system using available hardware have been identified by Chambers, et al. [25]:

(1) Open pressurizer PORVs,
(2) Open steam generator PORVs,
(3) Open reactor vessel head vent, and
(4) Isolate SI accumulators from the RCS.

Isolation of accumulators would be needed for RCS depressurization. However, the isolation of accumulators from the RCS may not be desirable from the accident prevention viewpoint. Although the additional water from the accumulators boiled in the core region would increase the primary system pressure and hydrogen generation in-vessel, it would also cool the core.

The operator actions identified in Reference 25 are not currently implemented in any PWR emergency guidelines. Chambers, et al., have suggested different bases for operator actions considering the following two factors:

1. The ready availability of thermal-hydraulic data and phenomenological considerations of the plant.

2. Ergonomic aspects of the layout and configuration of the control room environment.

The bases for the timing of the operator actions and the plant control room indicator for these actions are given in Table 3.5.

Table 3.5 Bases for Operator Actions and Indicators for RCS Depressurization (Reference 25)

Action/Indicator	Basis
Latch open pressurizer PORVs when steam generators dry.	All cooling lost, have waited for power recovery, operator action required.
Latch open pressurizer PORVs at five minutes.	Earliest time that operators would react based on plant dynamics and operations experts' "best guess".
Latch open reactor vessel head vent when boiling begins in core.	Added relief capacity needed for steam generation from boiling in core.
Latch open steam generator PORVs at 15 minutes.	Hypothesized as reasonable time for action based on plant operating philosophy and operations experts' "best guess".
Latch open steam generator PORVs at two minutes.	Hypothesized earliest time for action.
Steam generators dry.	All steam generator secondary side liquid levels indicated near top of tube sheet.
Boiling in core.	Subcooling margin = 0.0, core exit thermocouples at saturation temperature.
Isolate SI accumulators.	Bleed off nitrogen cover gas through manually opening air-operated valves.

These depressurization variations were tested in an analysis using the RELAP5 code [25]. The results show that for the transient sequence (TMLB'), a primary system pressure of 1.8 Mpa can be achieved in a scenario in which the pressurizer PORVs were opened when the steam generators were dry and the head vent was opened when boiling began in the core. The accumulators also were isolated. This system pressure was within the range of 1.5 to 2.5 Mpa (230 to 360 psia) that was assumed to be the DCH cutoff pressure range based on corium dispersal consideration [25].

The capability to effectively depressurize the RCS depends on the relief capacity of each plant. The studies performed for the Surry plant may not be applicable to other plants. The relief capacities of a group of PWRs are compared in Table 3.6 [26,27]. It is seen that many reactors, particularly those designed by Combustion Engineering and by Babcock & Wilcox, have smaller relief capacities than reactors designed by Westinghouse. For reactors with smaller relief capacity, only limited depressurization can be achieved through the actuation of PORVs. Therefore, using the PORV alone may not be sufficient to mitigate the potential for DCH in these reactors. It is also noted in Table 3.6, that five reactors designed by Combustion Engineering do not have PORVs.

Prior to the implementation of primary system depressurization procedures, the following must be considered for each plant:

1. Is depressurization technically feasible using existing systems or are modifications to the existing systems cost-effective?

Table 3.6 Summary of PWR Pressurizer Relief Capacities

Plant	No. of PORVs	Normalized[1] PORV Capacity (lb/hr-MW)
I. Westinghouse Plants		
Surry 1,2	2	86.0
North Anna 1,2	2	76.0
Beaver Valley 1,2	3	79.9
Millstone 3	2	61.3
Robinson	2	95.5
Trojan	2	61.6
Salem 1	2	63.0
Farley 2	2	79.2
Turkey Pt. 3,4	2	95.1
Prairie Island 1,2	2	108.5
Ginna	2	117.8
Point Beach 1,2	2	117.9
Zion 1,2	2	64.6
Haddam Neck	2	115.1
Indian Point 2,3	2	78.7
San Onofre 1	2	80.0
Kewaunee	2	106.0
Wolf Creek	2	61.6
Yankee Rowe	1	118.0
Braidwood	2	61.3
Byron 1 & 2	2	61.6
Callaway	2	61.6
Comanche Peak 1 & 2	2	61.6
Farley 1 & 2	2	79.2
Shearon Harris 1	3	75.4
South Texas 1 & 2	2	55.0
Vogtle 1 & 2	2	87.1
II. Combustion Engineering Plants		
Calvert Cliffs 1,2	2	56.7
Ark. Nuclear One 2	0	0
Fort Calhoun	2	69.7
Maine Yankee	2	57.0
Millstone 2	2	59.8
Palisades	2	60.5
St. Lucie	2	59.8
Waterford 3	0	0
San Onofre 2,3	0	0
Palo Verde 1,2,3	0	0
WNP-3	0	0
III. Babcock & Wilcox Plants		
Ark. Nuclear One 1	1	38.9
Oconee 1,2,3	1	41.7
Crystal River 3	1	40.8
Three Mile Island 1	1	38.9
Rancho Seco	1	40.4
Bellefonte 1,2	1	43.7
WNP-1	1	41.8
Davis-Besse	1	40.4

[1] Capacity of each PORV

2. Is sufficient information (instrumentation) available for the operator to determine the time for actuating primary system depressurization?

3. Is sufficient information (instrumentation) available for the operator to monitor the progression of depressurization and the condition of the reactor core?

4. Is the impact of depressurization on accident progression and potential risks fully understood?

3.3.2 Combustion Control

There are four combustion modes which could potentially occur in the containment: deflagration, diffusion flame, auto-ignition, and detonation. The pressure loading and thermal effects of the three former modes are not expected to threaten the containment integrity, because PWR containments are characterized by a large volume and high design pressure. It is the local detonation caused by non-uniform gas distribution, which could challenge containment integrity. Thus, combustion control in a PWR dry containment should focus on promoting gas mixing and deliberate burning in order to prevent the accumulation of combustible gases greater than the lean detonation limit.

A deliberate ignition system using igniters, such as glow plugs, has been adopted for PWR ice condenser and BWR Mark III containments. The use of glow plugs as the igniter system is based on their simplicity of operation (only requiring electrical power) and their compatibility with other instrumentation (no electrical noise from sparks). Glow plugs have been evaluated extensively at simulated LOCA conditions for reliability, endurance, and ignition performance. It appears that the system could also be used for PWR dry containments, if necessary. The installation of glow plugs (quantity and locations) is plant specific and depends on the containment size, internal structure layout and potential locations of release sources.

During an accident a large quantity of steam would be released from the reactor vessel to the containment. Because the PWR dry containment does not have any pressure suppression system, such as an ice-condenser or suppression pool, the containment is generally inerted if the containment sprays or fan coolers are not activated. Analyses have shown that combustion is not expected during many severe accident sequences because of steam inerting. However, these analysis do not extend long enough to include the natural condensation of steam from a cooling atmosphere and on cooling structures, which will eventually occur and hydrogen combustion may become a challenge. Combiners, even if available, were not designed for these conditions. A cost-benefit analysis of a hydrogen ignition system for the Zion and Surry plants [16], concluded that the installation of a full ignition system would not be cost effective. It is possible to reduce the threat from hydrogen combustion for both Zion and Surry, if one were to consider implementing additional operator procedures leading to operation of sprays (and containment fans, in the case of Zion) in such a manner that a certain steam fraction (10-20%) remains in the atmosphere indefinitely. An example of such a strategy (assuming the spray system is operating in the recirculation mode) was provided at the 19th Water Reactor Safety Information Meeting by a spokesman for Commonwealth Edison Company, the owners of the Zion plant: By throttling the valves controlling the flow of cooling water to the CSS heat exchangers, a hotter spray temperature is obtained, which results in a less rapid steam condensation, and thus provides a greater opportunity for steam concentration control.

3.3.3 Containment Venting

Venting has been suggested as a means by which an overpressure challenge to containment integrity could be mitigated, thereby reducing the risk of containment failure. The benefits of venting have been discussed for some BWR plants [28] and PWR ice-condenser plants [6]. The removal of radionuclides can be enhanced in an ice condenser plant by venting from the upper containment, and in a BWR plant by venting from the wetwell chamber (i.e., ice or water scrub the radioactive material). For a PWR dry containment, early venting will probably not be needed. Late venting to prevent late uncontrolled release of radioactivity due to containment structure failure may be beneficial but should be evaluated.

For PWR plants, a containment venting system is not considered a part of the engineered safety features and venting is not implemented in the EOPs. However, the containment purge system could be considered for limited and controlled venting following a loss-of-coolant LOCA. The containment purge system consists of air supply and exhaust parts and is designed for containment ventilation during normal operation.

The air exhaust system is equipped with an air duct, isolation valves, filters, and fans. The filters are composed of banks of prefilters and HEPA filters installed in series, and are arranged in two parallel independent modules. For the Zion plant each module has a rated flow of 20,000 cfm. The air fans are of the motor-driven type and are mounted in the outlet of the respective exhaust air filter. Two exhaust air fans are provided, each of 100% design capacity (40,000 cfm) with one fan as spare. It should be pointed out that the containment purge system is designed for operations limited to a maximum temperature of 120 F and pressures between -0.1 and 0.3 psig. The adequacy of this existing system for the purpose of containment venting during severe accidents must be fully evaluated.

In a PWR dry containment parametric study [29], the CONTAIN code was used for a preliminary calculation of containment venting for the Surry plant. The calculation was performed to investigate the possible benefit of opening a vent to relieve the slow pressurization during a small break LOCA sequence. A flow path with an area of 1 ft^2 to the outside environment was opened at 86,400 seconds (24 hours) and kept open until the problem end time of 90,100 seconds. The results show that this venting area is sufficient to terminate the pressurization at a pressure of 3.4 bar (49.3 psia). No discussion of the venting procedure with respect to the operator action, selecting of vent path, activation pressure, and equipment performance was given in Reference 29.

Some European countries (Sweden, Germany, and France) have installed, or intend to install, filtered vent systems for PWR plants. Strategies for containment venting to prevent overpressurization have been evaluated for severe accidents [30,31]. The strategies have also been applied to hydrogen control. For example, Bracht and Tiltmann [30] have performed an analysis for a large German PWR during core melt accidents. The study considered three strategies of containment venting for hydrogen control. A low-pressure loss of coolant sequence and a high-pressure station blackout transient were involved. Their results are summarized as follows:

(1) Early venting before the start of core heatup:
Early venting reduces the content of air in the containment prior to the in-vessel releases of hydrogen and steam. Thus, the containment atmosphere becomes more inerted and rich in hydrogen after core melt. The hydrogen enrichment may make the consequence of a potential hydrogen burn more severe.

(2) Venting during core melt:
Since most hydrogen is generated and released during the period of core melting, venting during this time would reduce the hydrogen content in the containment atmosphere. However, hydrogen enrichment in the vent system could make a deflagration or even a detonation possible if no countermeasures are provided. Another disadvantage of such a vent strategy is that venting occurs at a time when the fission product level in the containment is extremely high and aerosol settlement has just started.

(3) Late venting:
This strategy is applied when the containment pressure reaches the design pressure. The analyses show that late venting has no effect on the hydrogen situation in the containment while hydrogen enrichment in the vent system could lead to a hydrogen combustion.

3.3.4 ISLOCA Mitigation

In a detailed study of interfacing system LOCAs for PWRs [8], four corrective actions were identified that could be taken to reduce the core damage frequency (CDF):

(1) Application of continuous pressure (leak) monitoring devices;
(2) Increased frequency of valve leak testing;

(3) Improvement of operator training; and
(4) Implementation of RWST makeup procedure.

The first three actions are preventive and are capable of reducing the CDF due to ISLOCA by a factor of 1 to 6 depending on the specific plant.

Once an ISLOCA has occurred, the following actions could be implemented to mitigate its effects:

(1) Isolation of ISLOCA:
This action is often feasible because in many lines a number of valves exist which can be closed to compensate for the failure causing the ISLOCA. However, in order to isolate, the operator must be able to pinpoint the location of the break with a good deal of accuracy. This action could lead to an aggravation of the accident if operators open or close incorrect valves or shut down vital systems in their attempt to contain the leakage.

(2) RCS Depressurization:
A reduction of pressure in the RCS would reduce the mass flow rate at the break area. However, depressurization will cause rapid vaporization and early core uncovery which could lead to an early release of radioactive materials from the fuel. If the ECCS is available, then depressurization would allow the activation of the RHRS Pumps. The massive injection rate of the RHRS would enhance core recovery and mitigate the release of radioactive materials. In the event of ECCS failure, continued core degradation would lead to the relocation of molten core debris into the lower plenum. Under this situation, RCS depressurization could increase the probability of an in-vessel steam explosion due to the contact between core debris and water in the lower plenum.

As stated in Section 3.3.1, some PWR plants do not have PORVs to reduce the RCS pressure. This action will not apply to these plants.

(3) Refilling of RWST:
The RWST is the primary water supply for the ECCS. The capacity of a typical PWR four-loop plant is about 350,000 gallons (Zion and Surry). The water can last for about 1 hour at an injection rate of 6000 gpm (2 RHR pumps). Recirculation is normally provided by taking suction from the containment sump. However, the sump may not contain enough water during an ISLOCA event in which the break is outside of containment and water flowing out of the break cannot find its way back to the containment sump. Thus, it is crucial to provide actions to avoid or delay depletion of the RWST by adding makeup. The sources of water and capacities of pumps and lines available for RWST refill are plant specific and a review of some plant procedures has shown that many plants have existing procedures for refilling the RWST at limited rates once a low level in the tank is reached [9]. In many plants, existing capacities of pumps and lines for refill are likely to be inadequate for some important scenarios.

The effectiveness of this refill strategy depends on the makeup sources available at a specific plant and on the particular accident sequence taking place. For an ISLOCA sequence, recirculation of the auxiliary building sump water to the RWST could be accomplished utilizing the Radwaste System. The sump water would be processed through the Radwaste System, passed into the primary water system, and back to the RWST.

(4) Flooding the break location:
This action is aimed to provide fission production scrubbing. An alternate water source, such as service water, could be used if the break location can be identified and connections to the water system are available. One of the adverse effects is that flooding could impact the operation of equipment located near the site of the break.

(5) Auxiliary Building Fire sprays:
Activation of the fire sprays in the auxiliary building during an ISLOCA event could provide fission product scrubbing. This action may have an adverse impact on the operation of electrical equipment.

For an SGTR sequence, the following actions could be implemented:

(1) Closing main steamline isolation and bypass valves on the affected steam generator to isolate flow from the ruptured steam generator;

(2) Dumping steam from intact steam generators to initiate cooldown in the reactor coolant system; and

(3) Depressurizing the reactor coolant system to minimize the break flow.

The first action may cause the safety valves to open and discharge radioactive steam into the environment.

3.3.5 Reactor Cavity Flooding

Flooding the reactor cavity may provide debris coolability. A coolable debris configuration would eliminate or reduce the thermal attack of cavity concrete and therefore mitigate the containment overpressurization due to non-condensible gases. During an accident, if the ECCS and/or CSS are activated, it is likely ECCS and/or CSS cooling water as well as condensate would accumulate in the reactor cavity prior to the failure of the reactor vessel. The accumulation of water in the reactor cavity depends on the specific design of each plant. Some PWR plants do not provide a water path to the cavity and would not be able to collect water in the cavity. The presence of water in the reactor cavity before vessel failure has several effects on containment performance:

(1) For low-pressure sequences, the interaction between the discharging debris and water pool is likely to produce fine particles due to fragmentation of the core material. The large number of fine particles will increase the surface area for heat transfer which could result in a coolable debris configuration. On the other hand, a rapid generation of steam would cause a steam pressure spike but this is not expected to threaten the containment integrity of large-dry PWR plants as discussed in Section 3.2.5.

(2) For high-pressure sequences, water in the reactor cavity region could be co-dispersed with the core debris into the rest of the containment during a DCH event. The effect of water on DCH remains uncertain, as discussed in Section 3.2.2. Since CONTAIN analyses [13,14], as well as some experiments [40], have shown that co-dispersal of water with corium could increase the containment pressure during a DCH event, the benefit of applying this strategy for high-pressure sequences requires further confirmation.

In addition, the presence of water in the reactor cavity before vessel breach may result in steam explosions at vessel breach (Section 3.2.4). The adverse effect of this event has to be weighed against any expected beneficial effects.

Adding water to the reactor cavity after the vessel has breached and corium has discharged into the cavity is considered to be beneficial. An overlying water pool provides fission product scrubbing and enhances heat removal from the core debris. The increased debris coolability could arrest or delay basemat meltthrough.

At present, most of the PWR plants do not have specific systems for deliberate flooding of the cavity. However, deliberate flooding could be accomplished by using an alternate water source, such as fire water or service water systems, to directly fill up the cavity or cause an overflow from the containment floor to the cavity. For some PWR plants, the cavity is designed such that the curbs around the instrumentation tunnel could prevent water from flowing into the cavity. The implementation of reactor cavity flooding should consider the depletion of water resources which could be used for other purposes.

The Zion FSAR [32] reports that provisions have been made for the future installation of a post loss-of-coolant accident protection system (PLOCAP). A major component of the PLOCAP is the cavity flood subsystem. The primary function of the system is to provide a rapid initial injection of a large quantity of water into the vessel and vessel cavity following the failure of the reactor vessel. The suggested system contains cavity flood tanks, sumps, and pumps. Reactor cavity flooding is also being considered as a strategy to cool core debris in the vessel by submerging the bottom head. Whether this water level relative to the RPV is obtainable in a particular plant depends very much on reactor cavity design.

3.3.6 Utilization of Fire Water for Containment Sprays

Under accident conditions, containment sprays are often required to limit the containment pressure and temperature rise, and to enhance fission product removal. However, in some cases, the CSS may become unavailable either due to the failure of the spray pump during the injection mode or due to the failure of the recirculation system after RWST depletion. Under these situations, the fire water system could be considered as an alternate water source for containment spray operation. This strategy involves cross connecting the fire water pump discharge to the discharge lines of the containment spray pumps or RHR pumps by hard-piping connections or a temporary hose connection. For Zion the capacity of each of the two fire pumps is 2000 gpm at 125 psig to 140 psig, which is comparable to the CSS.

In addition to controlling containment pressure, temperature, and fission products, this strategy also provides cooling for core debris in the cavity region or on the containment floor, and can help to control hydrogen combustion. However, there are potential adverse effects as described below:

(1) This strategy is likely to dilute the boron concentration in the sump. Thus, implementation of this strategy before reactor vessel failure would raise concerns about recriticality when in-vessel injection is in the recirculation mode.

(2) The strategy would compete with other strategies which require the same water resources and equipment, such as flooding of the reactor cavity, a steam generator, or the auxiliary building.

(3) While this strategy is being implemented, fire fighting equipment may not be available for its intended purpose.

(4) Implementation of this strategy during the late phase of certain accident sequences could cause a rapid condensation of steam which in turn could de-inert the atmosphere and result in a detonable mixture as a large quantity of combustible gases may have already accumulated in the containment.

Finally, it should be pointed out a cross-connection of the fire water system may not involve the spray additive tank. Without sodium hydroxide (NaOH) in the water, the effect of sprays on iodine removal is reduced.

3.3.7 Fission Product Control

The objectives of this strategy are to reduce fission product releases. While some European countries have installed or intend to install filtered vent systems for PWRs with large dry containments, such a system has not been considered for domestic PWRs. Potential venting paths, which may exist in some specific domestic PWRs, are therefore not expected to provide significant filtering of fission products. Nevertheless, it may be possible to modify such specific vent paths so that they pass through a water filled volume which can provide some fission product scrubbing. However, the objectives for fission product control are often met by actions already taken to control the pressure and temperature of the containment atmosphere. The actions are summarized below:

(1) CSS operation can scrub fission products released to the containment atmosphere. The fire water system can be used as an alternate source when the normal CSS water source is unavailable.

(2) Flooding the reactor cavity after vessel breach could cool the hot debris to reduce fission product volatilization. The overlying water pool aids fission product scrubbing.

(3) RCFCS operation will cool the containment. The cooling coil and filters of the fan system enhance the removal of aerosols.

(4) Flooding a steam generator with water following an SGTR, and flooding the break location following an ISLOCA can provide fission product scrubbing.

(5) Operating fire sprays in the auxiliary building following ISLOCA to provide fission product scrubbing and reduce fission product revaporization.

The benefits and potential adverse effects of these actions have been discussed in previous sections.

3.4 Strategy Implementation

An effective implementation of the CRM strategies requires several action steps. The first action involves an assessment of the plant damage conditions. These conditions provide a means to characterize the extent of core and containment damage, and the extent of fission product release. Oehlberg, et al., [6] have discussed the assessment of both RCS and containment damage conditions with respect to the implementation of Accident Management Guidance (AMG). They have characterized the containment damage conditions into four categories according to the status of containment function:

(1) The containment is closed and no existing challenge exists to containment integrity. Core debris, if any is being cooled, combustible gas concentrations are within acceptable limits and containment heat removal is available.

(2) The containment is intact, but threatened by lack of cooling or combustible gases. The containment function is still being performed, but loss of the function could occur unless physical conditions in the containment are improved. This condition includes both cases with core-on-the-floor and not.

(3) The containment is impaired in some manner and the containment function is lost, but not bypassed. As such, fission product release from the containment may still be mitigated by measures such as spraying the containment atmosphere or debris cooling.

(4) The containment is bypassed releasing fission products directly from the core debris to outside the containment. As such, measures to control fission product release from the containment such as spraying containment atmospheric or debris cooling may not be effective. However, the use of fire sprays in the secondary containment/auxiliary building, or submergence of the break could be effective in mitigating the release.

The four categories can be used to specify the containment conditions at any given time in the accident progression. These conditions could be indicated by a range of plant parameters. Examples of containment damage conditions and possible indicators given in Reference 6 are shown in Table 3.7.

The second action requires the full utilization of available plant information sources to direct operator actions. This involves both the safety related and non-safety related instrumentation. Instruments which are not qualified for severe accident conditions can often provide useful information as a backup to verify plant status. In addition, the failure of an information source may in itself provide valuable information. An example which relates the containment temperature and pressure control with plant information sources given in Reference 6 is shown in Table 3.8. The table shows a number of alternative means for making determinations regarding containment pressure, H_2 levels, temperature, and spray status. For example, the operability of air operated equipment, such as primary system relief valves, could indicate containment pressure. The opening of primary system relief valves generally requires a differential pressure between the instrument air system and the containment atmosphere.

The third step requires the estimate of plant response after a CRM action has been implemented. This action may involve calculations and interpretation of the plant states from plant information sources. The estimate of plant response should be used to judge the effectiveness of the strategy and predict the future plant conditions.

Table 3.7 Example Containment Damage Conditions and Possible Indicators* (Reference 6)

Containment Damage Condition	Possible Indicators
Containment Closed and Cooled	• Isolation complete • Containment pressure and/or temperature stable/decreasing and within the design basis • Containment hydrogen levels stable and less than 4% • Containment heat removal functioning normally • Sufficient water addition to containment to match decay heat removal requirements
Containment Threatened	• Isolation complete • Containment pressure and temperature increasing • No fission products detected outside containment • Containment hydrogen levels high and/or increasing • Containment heat removal function possibly impaired • CO and/or CO_2 levels in containment high and/or increasing • Insufficient water addition to containment to match decay heat removal requirements
Containment Impaired (Includes Failed)	• Isolation not complete • High radiation levels in secondary containment/auxiliary building • High temperatures in secondary containment/auxiliary building • High humidity in secondary containment/auxiliary building • Containment pressure decreasing without sufficient heat removal • High radiation levels outside containment • Fire alarms or fire system actuation in secondary containment/auxiliary building with no signs of fire
Containment Bypassed	• High radiation, temperatures and/or humidity levels in secondary containment/auxiliary building while containment conditions remain "normal" • Increasing sump levels in secondary containment/auxiliary building • Containment pressure decreasing without sufficient heat removal • High radiation levels outside containment • Fire alarms or fire system actuation in secondary containment/auxiliary building without signs of a fire

*based on instrument readings and/or analysis estimates

Table 3.8 An Example of the Relationship of Safety Functions to Plant Parameters and Information Sources (Reference 6)

Safety Function	Potential Plant Parameters	Example Plant Information Sources	Example Plant Systems Supporting Safety Function
Containment Temperature and Pressure Control	- Containment Pressure	- Control Room Pressure Indicators - Local ILRT Pressure Indicators - RCS/RPV Pressure Indicators - Pressure Indicators on Inactive Pumps Aligned to the Containment Sump or Suppression Pool - Containment Air Operated Component Operability - Shutoff Head/Flow of Containment Injection Sources	- Containment Sprays - Containment Air Coolers - Containment H_2 Recombiners - RCS Injection Sources - Recirculation Spray System - RHR System - Igniters (Ice Condensers) - H_2 [illegible] System - Containment Purge System - RHR System
	- Containment H_2 Levels	- Control Room H_2 Monitors - Post Accident Sampling Results	
	- Containment Temperature	- Control Room Containment Temperature Indicators - Control Room Containment Ventilation System Temperature Indicators (numerous systems and indicators exist)	
	- Containment Spray Status	- Control Room Containment Spray Flow Indicators - Local Containment Spray Pump Suction Pressure Indicator	

4 Applications to Selected Sequences

In this section the strategies are assessed by applying them to several accident sequences, which have been identified as major contributors to plant damage [2]. Containment analysis performed using the STCP, MELCOR, CONTAIN, and MAAP codes are used for discussion. The Zion and Surry plants were used as the surrogate plants for this assessment.

4.1 Station Blackout Sequences

Station blackout sequences are initiated by a loss of all off-site and on-site ac power, which results in the unavailability of the high-pressure injection system, the containment spray system (both injection and recirculation modes), the containment fan cooler system and the auxiliary feedwater pumps. While the loss of all ac power does not affect instrumentation at the start of the station blackout, a long duration station blackout leads to battery depletion and subsequent loss of vital instrumentation. Battery depletion was postulated to occur after approximately 4 hours for Surry, and 6 hours for Zion [2]. Thus, the most important accident management activities after a station blackout should be (1) to recover the ac power, (2) to extend dc power, and (3) to identify and utilize alternate systems and resources for cooling of the core and containment. If recovery and mitigative actions are not successful, the event will result in core degradation, vessel breach, and eventual containment failure.

The NUREG-1150 study [2] concluded that a station blackout event could lead to (1) a reactor coolant pump seal LOCA due to the loss of all seal cooling, and (2) a SGTR induced by high temperatures due to natural circulation of hydrogen, steam, and fission products in the primary coolant system. These two scenarios will be discussed in Sections 4.2 and 4.5, respectively. The scenario discussed in this section assumes no pump seal LOCA and no SGTR.

4.1.1 Containment Response

At the initiation of the accident, the main coolant pumps stop and the reactor trips. Following the reactor trip, decay heat boils off the water in the secondary side of the steam generators. After steam generator dryout, the primary system pressure rises to the relief valve setpoint and reactor coolant is discharged in a cycling manner to the quench tank and ultimately to the containment building. The failure of high pressure injection (HPI) to provide coolant makeup causes core uncovery, fuel heatup, melting, and eventual vessel breach at high pressure. Because the station blackout considered in this scenario is a high-pressure sequence, prior to vessel breach most of the steam, hydrogen, and fission products generated during core degradation are retained in the RCS. Only a relatively small quantity is released to the containment through the cyclic opening of the PORV. This amount has no significant impact on the large-dry PWR containment. At the vessel breach, molten core debris is ejected under high pressure to the reactor cavity. The rapid depressurization in the RCS will activate the accumulators and add water to the reactor cavity. The corium-water and corium-concrete interactions will produce a large quantity of steam and noncondensible gases which could cause over-pressurization and over-heating in the containment. A potential scenario associated with the high pressure melt ejection is the DCH event which would result in a different containment response. The DCH event will be discussed in Section 4.2.

The above station blackout sequence was analyzed for both the Zion and Surry plants using the STCP code [7, 33, 34]. The predicted major events are [7]:

	Zion	Surry
Steam Generator Dryout	97 min	80 min
Start of Core Melt	148 min	135 min
Reactor Vessel Breach	188 min	177 min
Containment Failure	36 hr[1]	72 hr[2]

[1] extrapolated time for over-pressurization failure
[2] extrapolated time for basemat meltthrough

A loss of dc power was assumed at the initiation of the accident in the above analyses. Because both plants have batteries as the standby dc power sources, core damage and containment failure predicted by the MARCH code would be delayed by 4 to 6 hours, if the batteries are maintained fully charged. This delay arises because most plants have at least one turbine powered auxiliary feedwater/pump which can be operated if dc power is available.

4.1.2 Containment Challenges

4.1.2.1 Zion plant

Figure 4.1 shows the containment pressure and temperature for the SBO sequence as predicted by the STCP code [7]. Immediately after the reactor vessel breach (188 minutes), the corium/water interaction yielded a steam pressure spike of about 17 psig, which did not threaten the containment integrity. However, the containment was continuously pressurized as a result of the release of non-condensible gases from corium-concrete interaction and the generation of steam from the overlying water pool. At about 550 minutes, the cavity water was depleted and the containment pressurization was solely due to the non-condensibles. The pressurization rate was reduced to 0.03 psi/min. At this rate, the containment is predicted to fail at about 2100 minutes (36 hours). Figure 4.1 also illustrates that the containment temperature was maintained at about 320 F throughout the transient. Although the temperature is above design temperature (270 F), it is unlikely to have a severe effect on containment integrity.

The STCP code also calculated the cavity concrete erosion rate as about 0.087 cm/min (Figure 4.2). At this erosion rate, the complete penetration of the containment basemat (2.74 m) would take more than 50 hours. Figure 4.3 shows the mole fractions of gases in the containment. The steam fraction is nearly 0.8 and hydrogen is below 0.04. The containment is highly inerted and no combustion was predicted by the code.

Although there are uncertainties in the STCP calculation, the general behavior predicted by STCP is similar to that predicted by other codes, such as MAAP and CONTAIN. It appears that overpressurization is an important long-term challenge to the Zion containment integrity.

4.1.2.2 Surry plant

For the subatmospheric plant, the containment pressure was 10 psia prior to the start of accident. The containment pressure reached 45 psia immediately after vessel breach at about 177 minutes as shown in Figure 4.4. Due to the specific design of the Surry plant, the reactor cavity is isolated from the containment sump and no water can flow back to the reactor cavity. The boil-off of the initial cavity water ejected from the reactor vessel at the time of vessel breach, increased the containment pressure to a peak of 55 psia. After the depletion of cavity water, the containment pressure increase is solely due to the non-condensible gases from the corium-concrete interaction. Figure 4.4 shows that the pressurization rate is only about 0.018 psi/min. At the end of 24 hours, the pressure reached 63 psia, which is slightly above the containment design pressure (60 psia) but considerably less than the estimated capacity (143 psia). An extrapolation of the containment pressure shows that the containment would fail at about 90 hours.

The containment temperature, included in Figure 4.4, is maintained at about 260 F during the entire transient. Although the temperature is above the design level (150 F), significant leakage due to the degradation of the containment penetration seals is not expected at this temperature level.

The molar fractions of steam, hydrogen, and oxygen are given in Figure 4.5. Although the hydrogen fraction is above 0.08 (the flammability limit), no combustion was predicted due to the high fraction of steam (above 0.55).

The concrete axial erosion under a "dry-cavity" situation is shown in Figure 4.6. The erosion rate is about 0.068 cm/min. At this rate, a complete erosion of the basemat (about 3 m) would take about 72 hours. It appears that for Surry, basemat meltthrough occurs prior to containment failure due to gradual overpressurization.

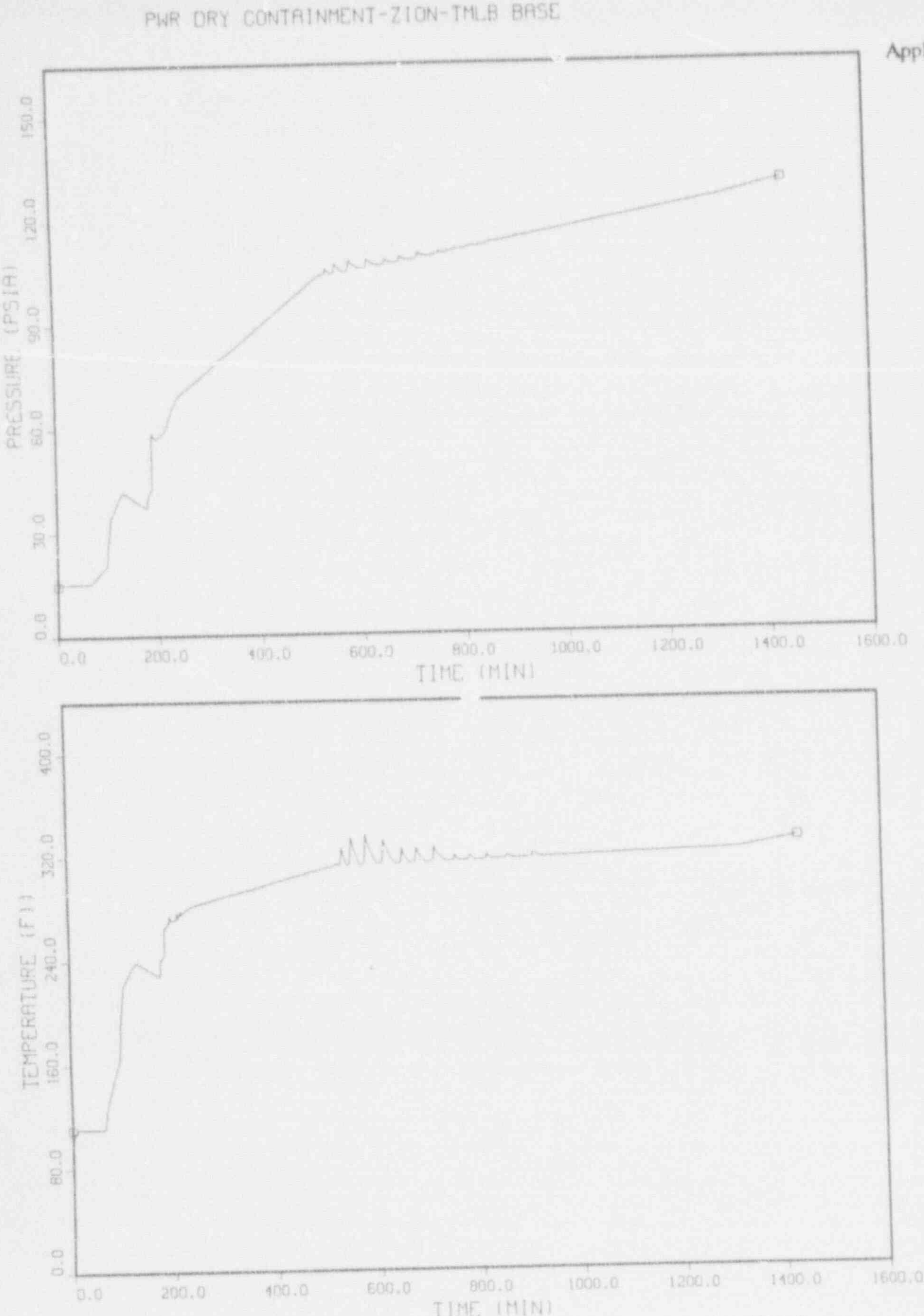


Figure 4.1 Containment Pressure and Temperature for Zion SBO Sequence

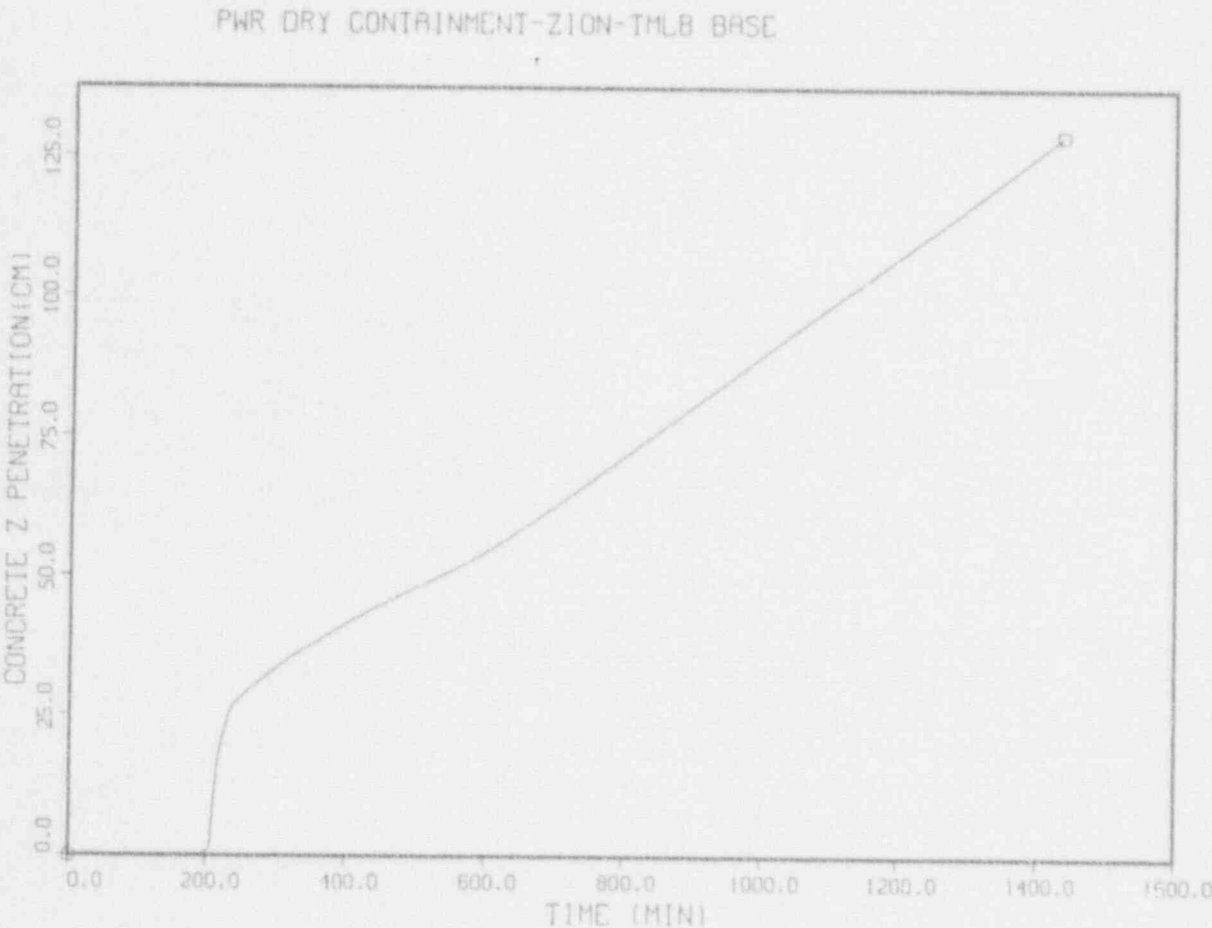


Figure 4.2 Cavity Concrete Erosion for Zion SBO Sequence

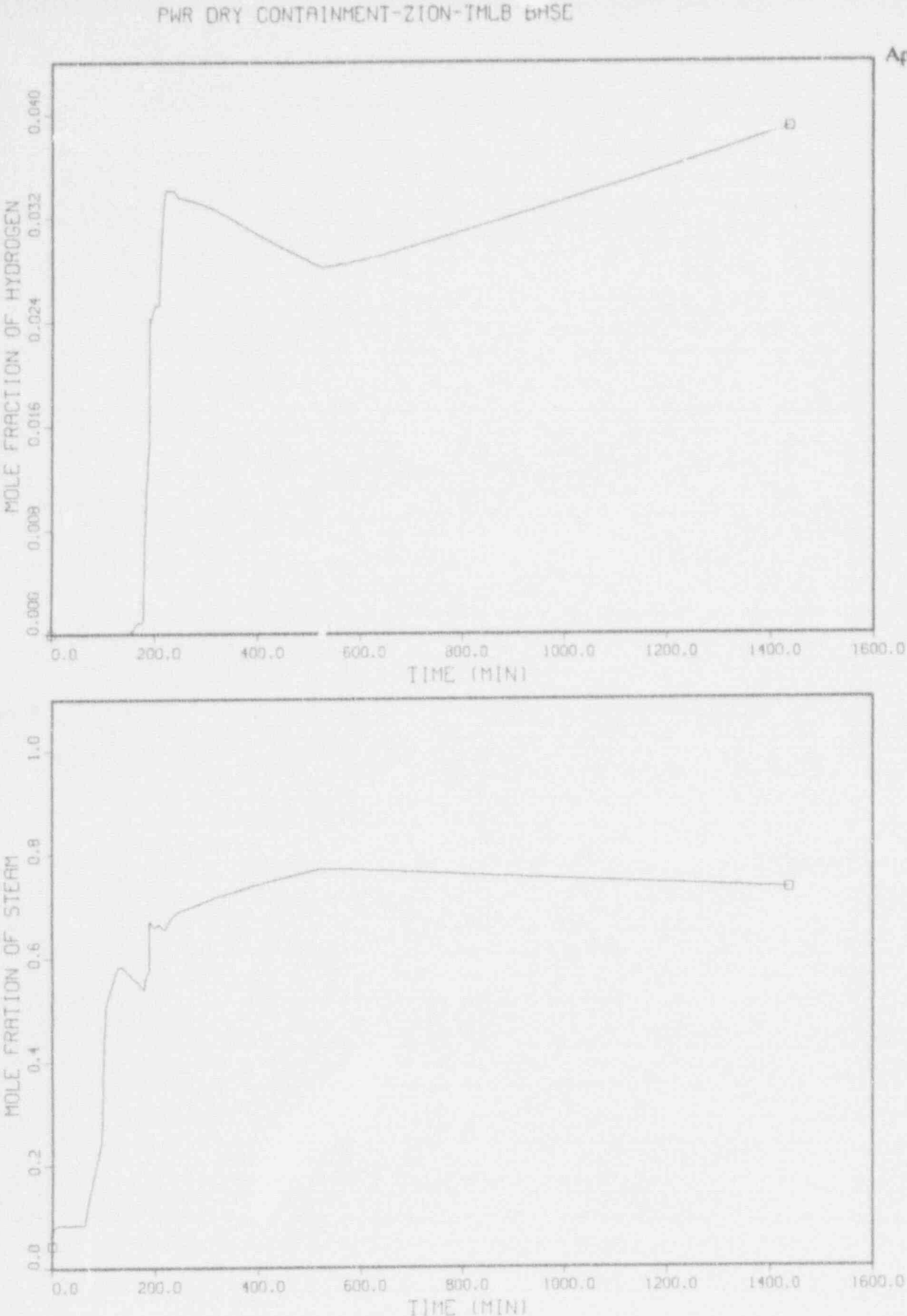


Figure 4.3 Gas Mole Fractions for Zion SBO Sequence

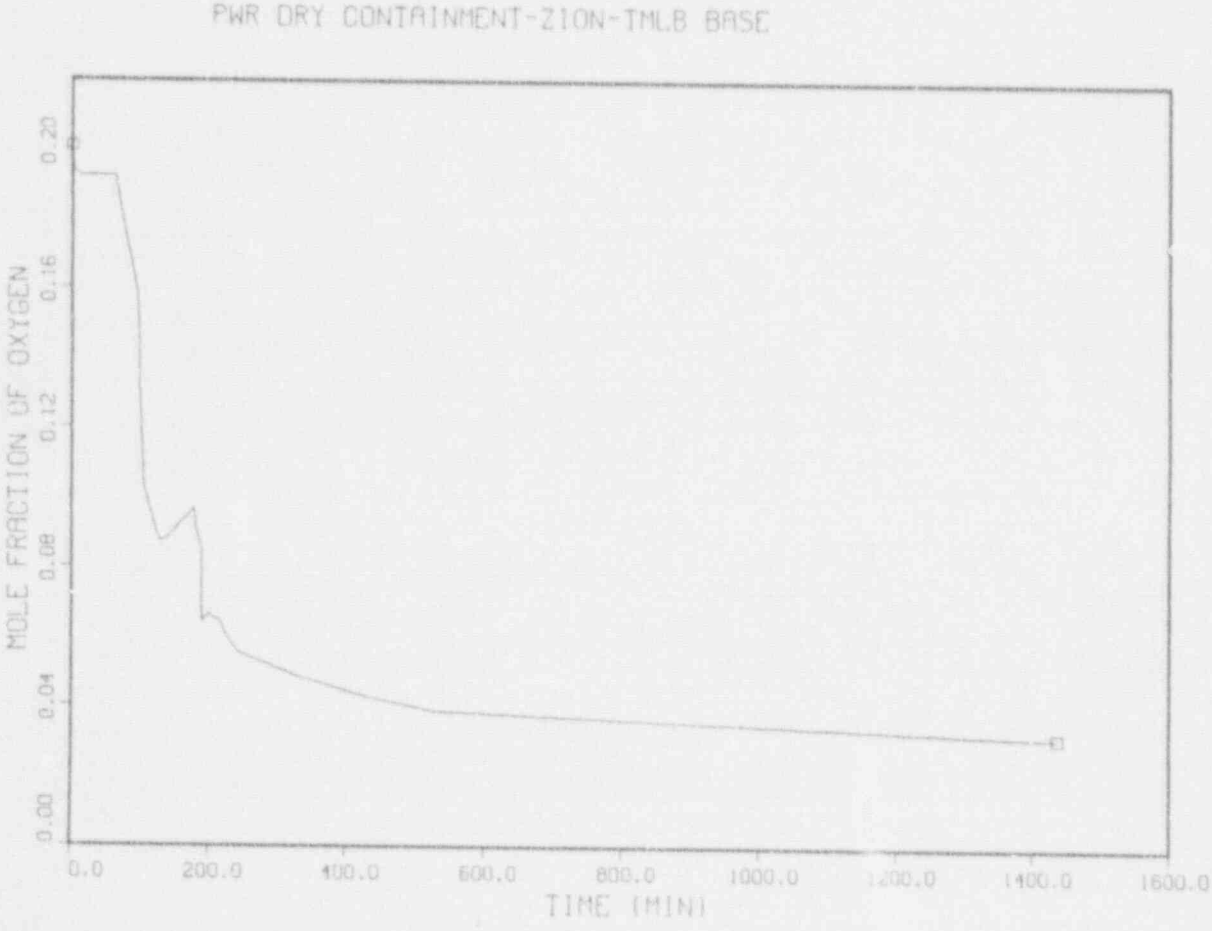


Figure 4.3 Gas Mole Fractions for Zion SBO Sequence (Continued)

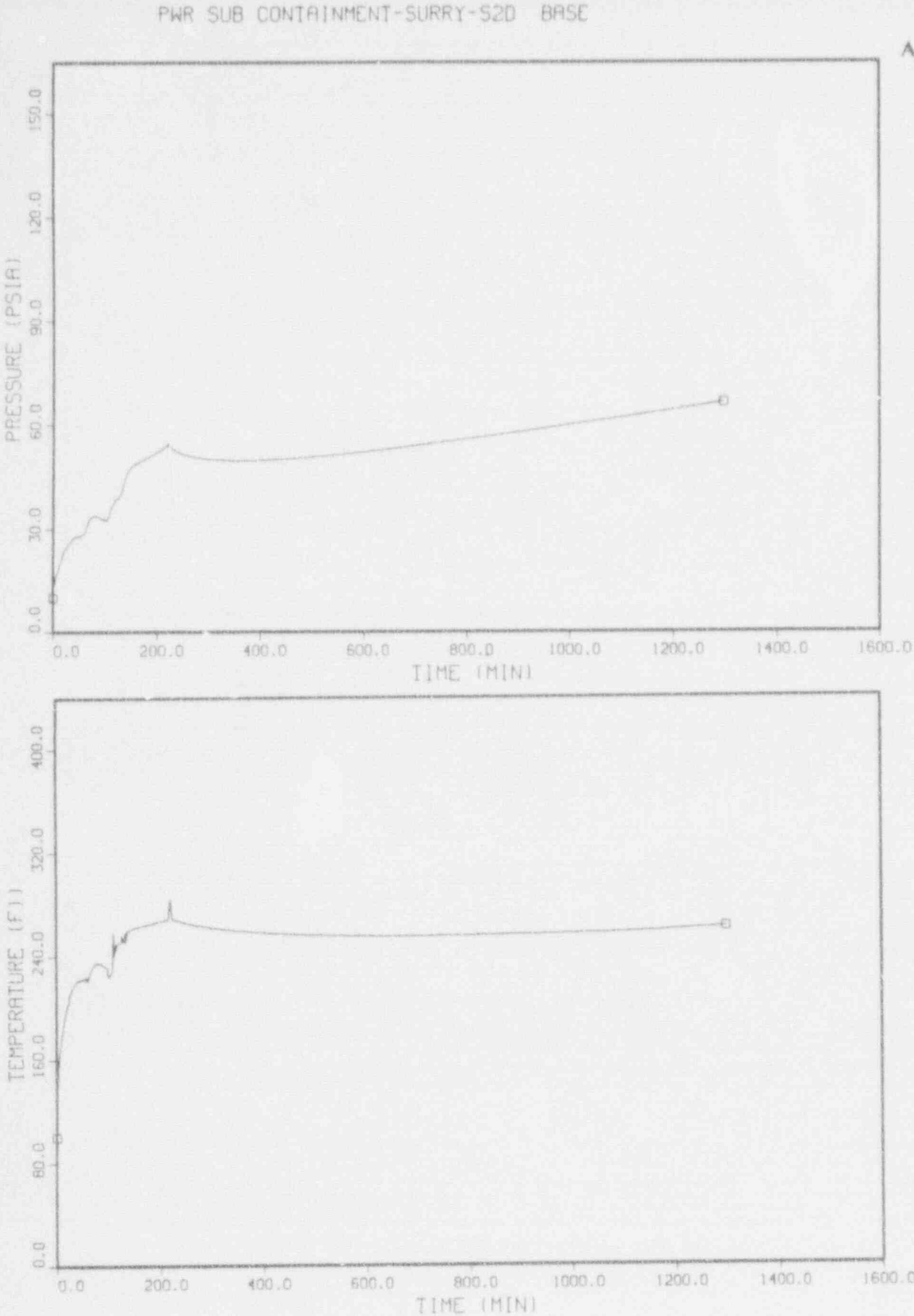


Figure 4.4 Containment Pressure and Temperature for Surry SBO Sequence

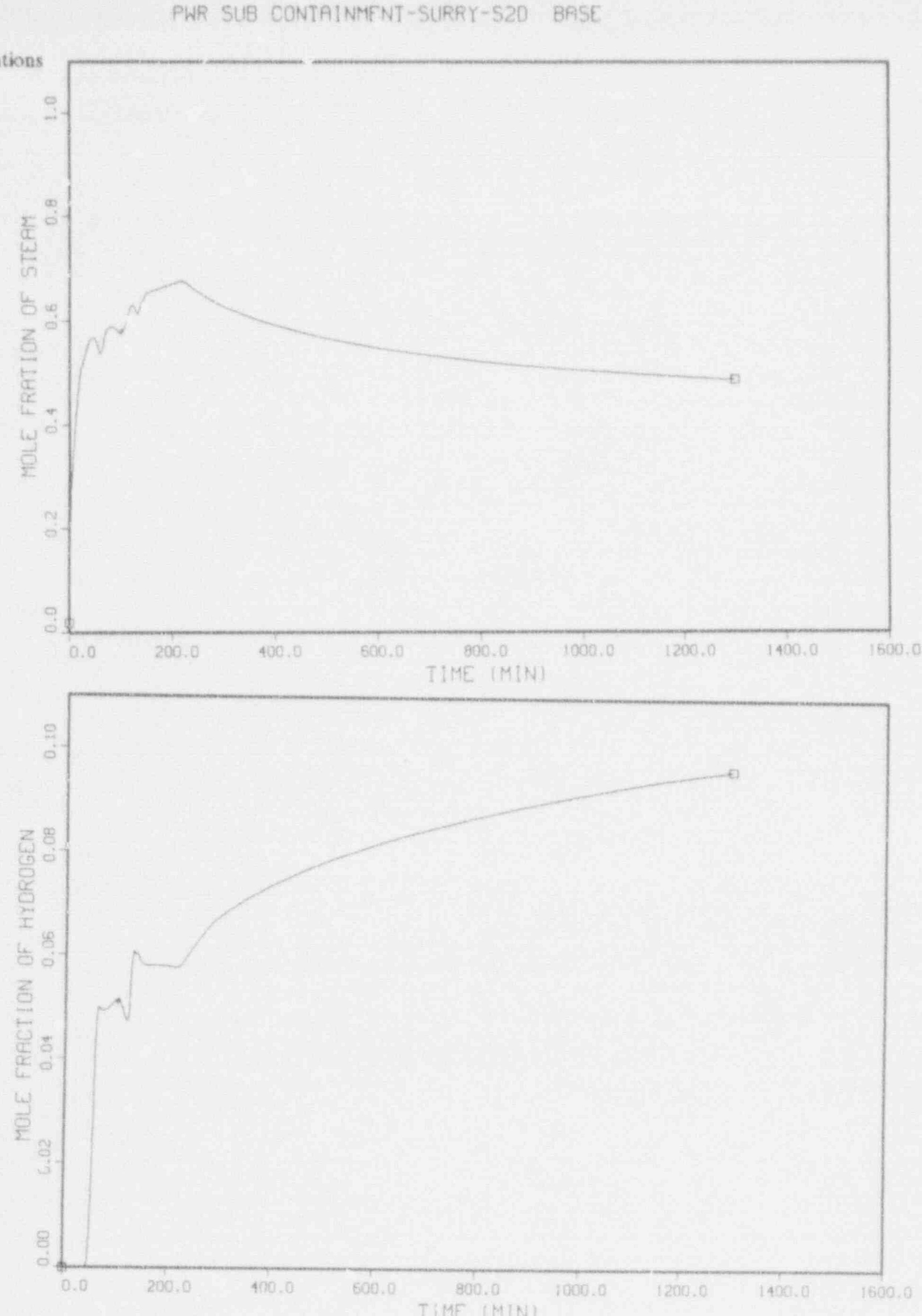


Figure 4.5 Gas Mole Fractions for Surry SBO Sequence

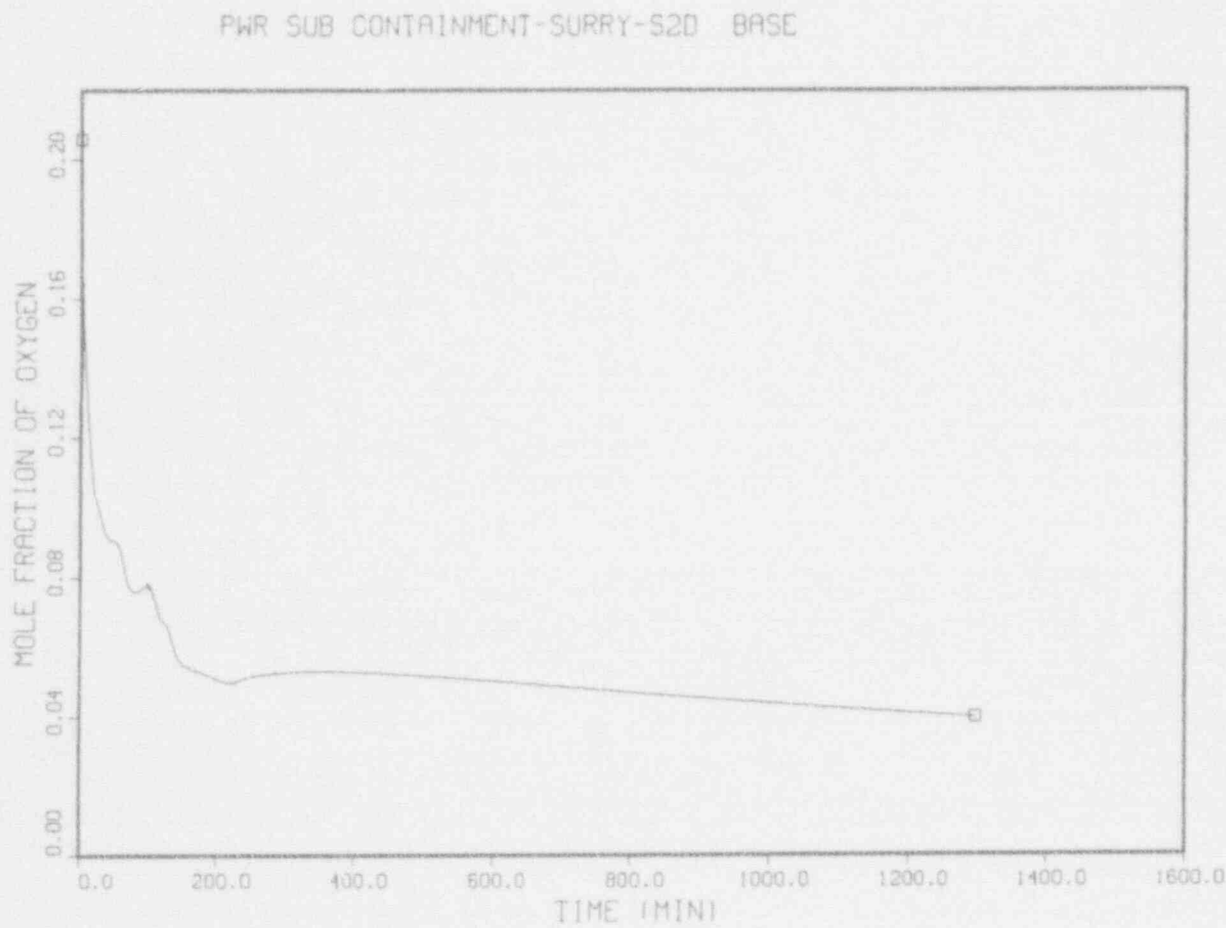


Figure 4.5 Gas Mole Fractions for Surry SBO Sequence (Continued)

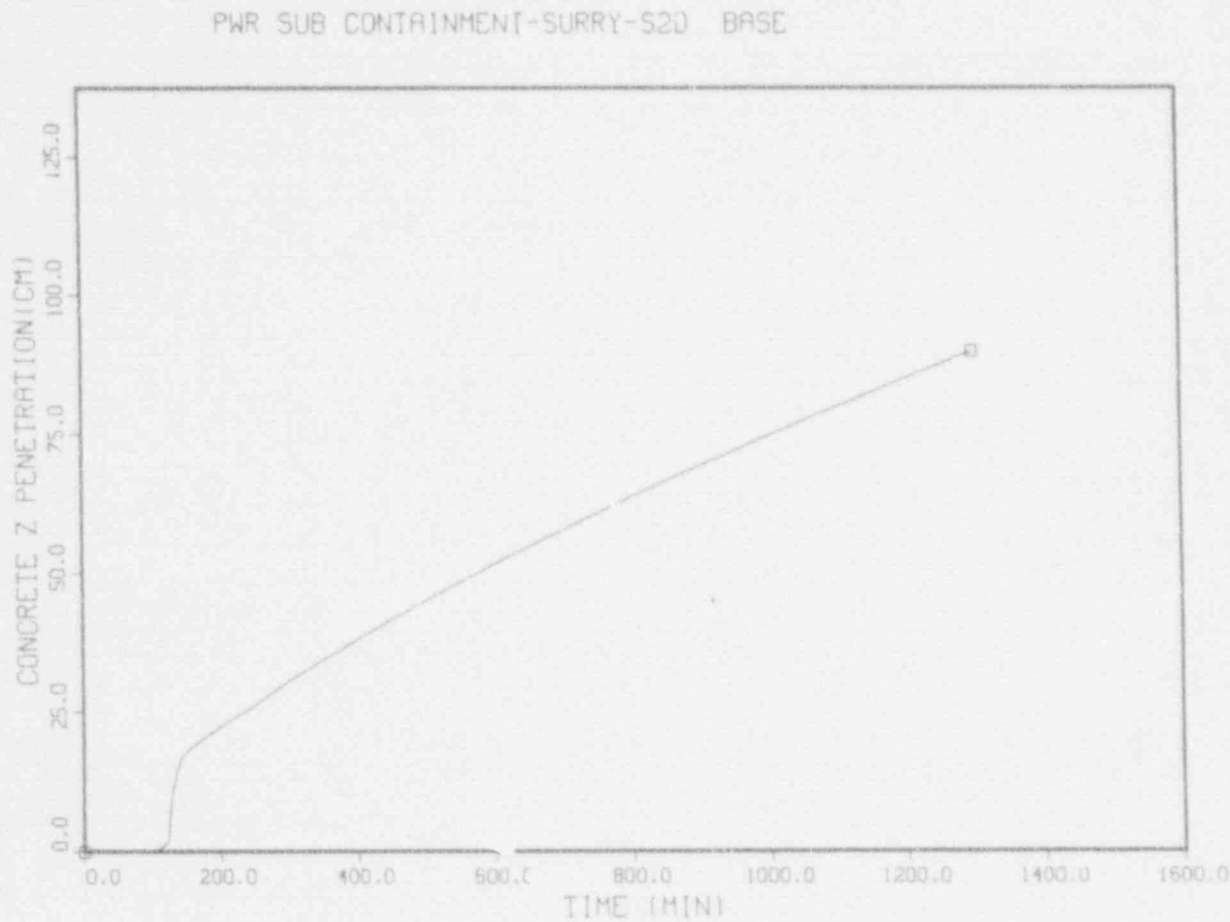


Figure 4.6 Cavity Concrete Erosion for Surry SBO Sequence

4.1.2.3 Implications

From the above calculations, it is clear that the presence of water in the reactor cavity has a significant impact on the long term challenges experienced by large volume containments. With water present in the reactor cavity, pressurization of the containment via steam and noncondensible gas generation will be rapid. Under these circumstances basemat penetration is unlikely to occur. If basemat penetration does happen, it will occur a long time after failure of the containment via overpressurization. On the other hand, if the cavity is dry, the steam and noncondensible gases released during CCI are unlikely to cause an overpressurization failure of a large dry containment. Under these circumstances, basemat failure or failure of seals due to high temperature are the most likely cause of loss of containment integrity.

4.1.3 CRM Strategies

Loss of off-site power and of all on-site AC power are sufficient conditions for declaration of an Alert. This condition is upgraded to a Site Area Emergency when SBO persists for more than 15 minutes. Among the licensee actions required by a Site Area Emergency are [35]:

- prompt notification of the state or local off-site authorities of emergency status,
- activation of on-site Technical Support Center, on-site Operational Support Center, and near-site Emergency Operations Facility,
- dispatch of on-site and off-site monitoring teams,
- making senior technical and management staff on-site available for periodic consultation with NRC and state authorities,
- providing meteorological data and dose estimates to off-site authorities.

According to Westinghouse ERGs, the operator is directed to the Emergency Contingency Action ECA-0.0, which is specific to station blackout. Some of the actions recommended in ECA-0.0 are listed below:

- attempt to emergency start the diesels,
- call for dedicated AC power
- attempt to switch around faulted circuitry at the switchyard,
- shed non-essential DC loads,
- attempt to manually align and set up the turbine-driven AFW pump,
- locally isolate RCP seal lines,
- locally open SG PORVs,
- manually effect containment isolation.

All the previous actions, except effecting containment isolation and shedding non-essential DC loads, are assumed to be ineffective. In-vessel accident management strategies to be initiated in the early phase of the accident are expected to center around attempts to depressurize the steam generators so that diesel-driven fire water pumps, if available, may be utilized to flood the steam generators and bring about some degree of core cooling. The failure of this in-vessel strategy would lead to eventual vessel breach.

The mitigation strategy during the late phase of the accident, i.e., after vessel breach, is plant-specific. Discussions for the Zion plant and the sub-atmospheric Surry plant are given below:

4.1.3.1 Zion plant

Since long-term over-pressurization appears to be a very important challenge to the Zion plant, mitigation strategies should focus on restoration of containment sprays and fans or containment venting. If venting is attempted, it should not be initiated immediately after vessel breach when containment pressure is rapidly increasing to the design pressure. The puff-release of fission products from the reactor vessel to the containment at the time of vessel breach is likely to cause a large release to the environment if venting is implemented at this time. A delay of venting besides allowing more decay of radioactivity, would provide sufficient time to settle aerosols released from the reactor vessel and from the corium-concrete interaction. The times of maximum CCI will depend on the scenario, especially the amount of water, if any, in the cavity. For the Zion SBO sequence of Figure 4.1, BNL has performed a venting analysis using the STCP code. In the analysis, venting is actuated at 370 minutes when the containment pressure reaches 85 psia. The equivalent diameter of the venting area is 1 ft. The vent valve is assumed to stay open without re-closing at lower pressures. The venting is actuated at about 3 hours after the vessel breach. It is expected that opening of the vent valve at this time will cause a less severe release of fission products than earlier valve actuation. Figure 4.7 illustrates the impact of venting on containment pressure and hydrogen mass. Within about 200 minutes, the containment pressure is reduced to the atmospheric pressure level and the quantity of hydrogen is reduced from about 1800 pounds to 300 pounds. At the end of 1450 minutes, the containment holds about 900 pounds of hydrogen, which is considerably less than the 3300 pounds that would be present if venting was not activated. Although there is a large reduction of hydrogen mass, the volume concentration of hydrogen increases from 4% to about 8% at the end of 1450 minutes, as shown in Figure 4.8. This is due to the simultaneous loss of steam and air by venting. The high concentration of hydrogen (8%) does not induce any combustion because of the lack of oxygen in the containment due to venting. The oxygen volume concentration is below 2% as shown in Figure 4.8. It should be noted that the implementation of a venting system could involve extensive hardware changes for most PWR dry containments. The valves of the containment vent and purge system are usually large butterfly valves and are unlikely to open once a

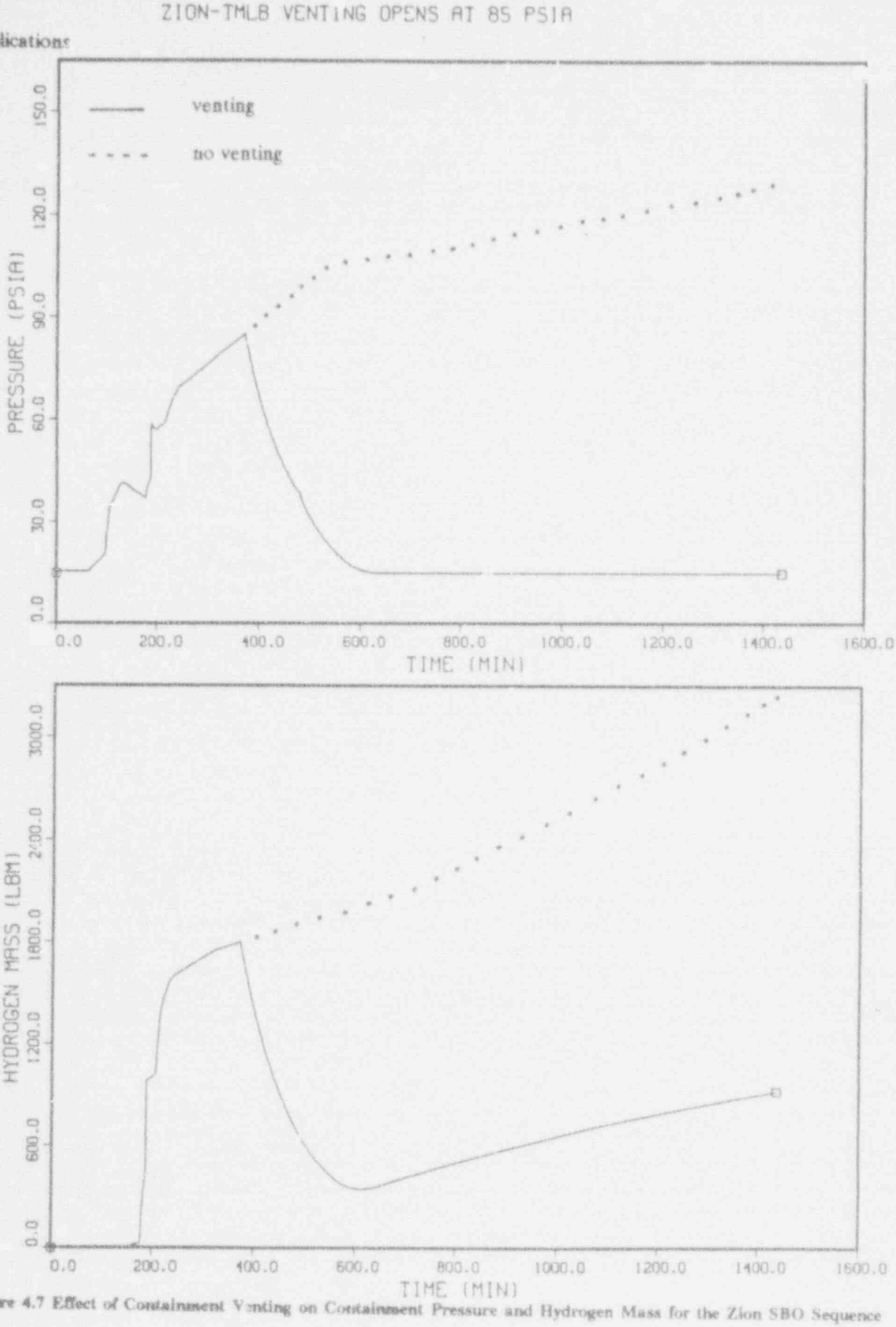


Figure 4.7 Effect of Containment Venting on Containment Pressure and Hydrogen Mass for the Zion SBO Sequence

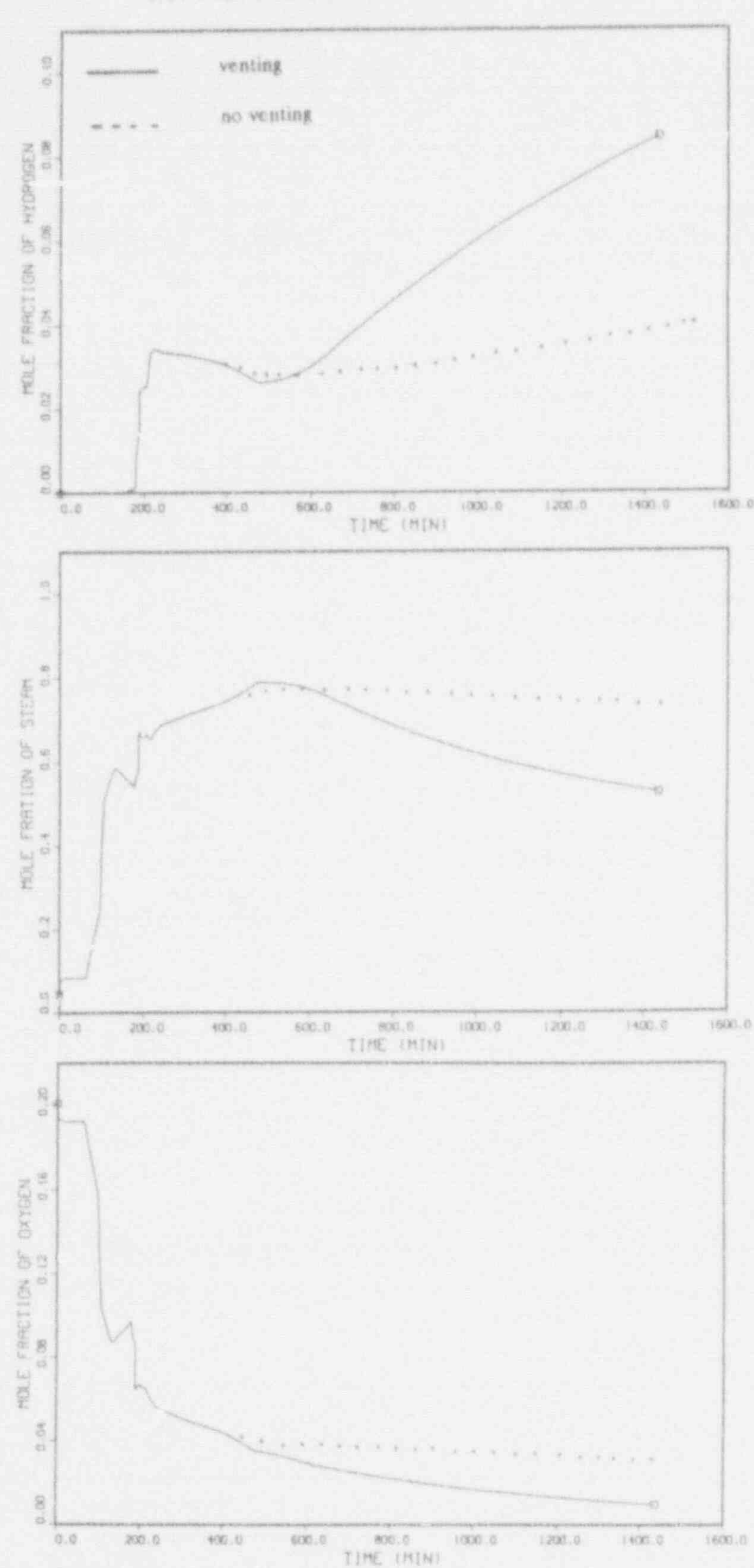


Figure 4.8 Effect of Containment Venting on Gas Compositios in Containment for the Zion SBO Sequence

significant pressure difference develops. Vent paths of a smaller diameter, capable of sustaining higher pressures, may exist or be relatively easily constructed from existing penetrations on a plant specific basis. However, providing a remote valve opening capability, especially one available during station blackout, would still involve major modifications.

The restoration of containment sprays and fans can also prevent or slow down the over-pressurization of the containment. A limited STCP analysis was performed for the Zion SBO sequences [7]. A summary of the results is given in Table 4.1. Restoration of containment sprays was considered for two cases. In one case, the sprays were restored at the time of vessel breach, and in the other case at about 4 hours after the vessel breach. In both cases, recirculation was assumed, so that sprays were continued throughout the transient after their initiation. The containment sprays provided cooling of the atmosphere and a water source to flood the reactor cavity. However, for these sequences, heat exchangers were not available to cool the circulating water, and soon saturated water was recirculated through the sprays. The erosion of cavity concrete was reduced considerably, but the corium was not quenched by the overlying water pool. Consequently, the corium/water interaction in the cavity provides a continuous steam source to the containment atmosphere, which causes a slow but continuous pressure increase as shown in Figure 4.9. Thus, the operation of containment sprays could delay, but not eliminate, the containment failure due to over-pressurization. Another consequence of corium interaction with the overlying water pool is that the steam generation maintained the containment inert. Combustion is unlikely to threaten the containment integrity.

Table 4.1 Summary of MARCH Analysis for the Zion TMLB' Results

Case	Description	Containment Pressure @24 Hr. psia	Concrete Erosion @24 Hr. cm	H_2 Burn	Containment Failure* (Time, min)
BS	No operator action.	128	125	No	No
S1	Sprays started at 200 min.	107	80	No	No
S2	Sprays started at 480 min.	106	80	No	No
F1	Four fan coolers started at 300 min. No sprays.	107	80	Yes	No
F2	Four fan coolers started at 960 min. No sprays.	CF*	75	Yes	Yes (1056)
F3	Two fan coolers started at 960 min. No sprays.	CF*	90	Yes	Yes (1207)

*Containment failure at 149 psia due to late H_2 burn.

The effect of fan coolers is quite different from that of sprays. The Zion plant has five containment fan cooler units. The heat removal capacity for each unit is about 24 MW (81 x 10^6 Btu/hr), which is comparable to the decay power (32 Mw) at the time of vessel breach. Hence, an operation of 2 or more fan units would be sufficient to remove the decay power and mitigate the pressurization effect. However, it should be noted that the dehumidification achieved by the fan cooler will deinert the containment atmosphere and could initiate combustion if ignition sources are available. An STCP calculation [7] shows that a late restoration of fan coolers could cause the burn of a large quantity of hydrogen and fail the containment as shown in Figure 4.10 and in Table 4.1. Therefore, application of this strategy during the late phase of the transient must proceed with caution. Monitoring of local concentrations of combustible gases (H_2 and CO) and inertants (steam and CO_2) should be performed to ensure that deinerting of containment atmosphere will not cause local detonation or the general deflagration of a large quantity of combustible

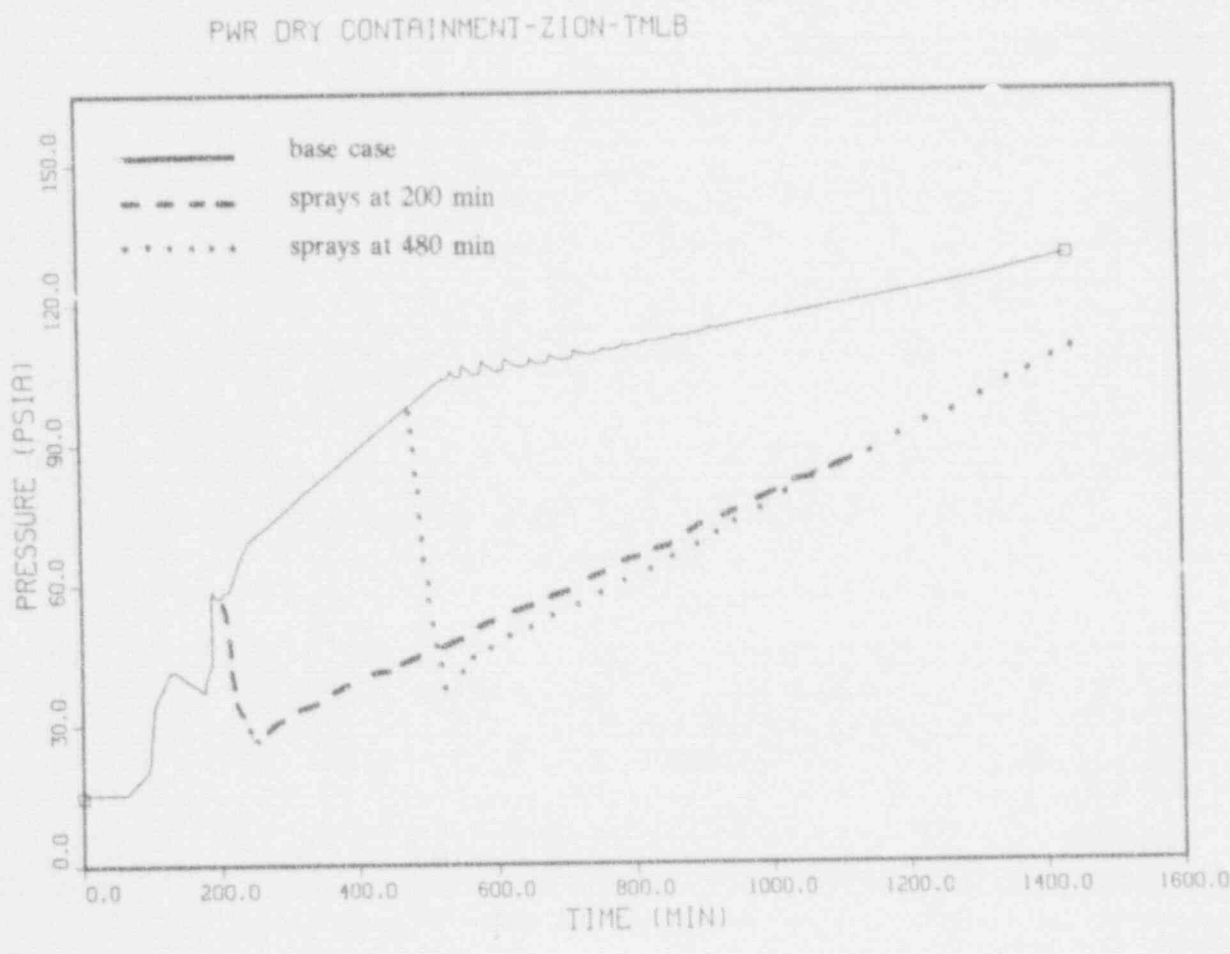


Figure 4.9 Effect of Containment Sprays on Pressure for Zion SBO Sequence

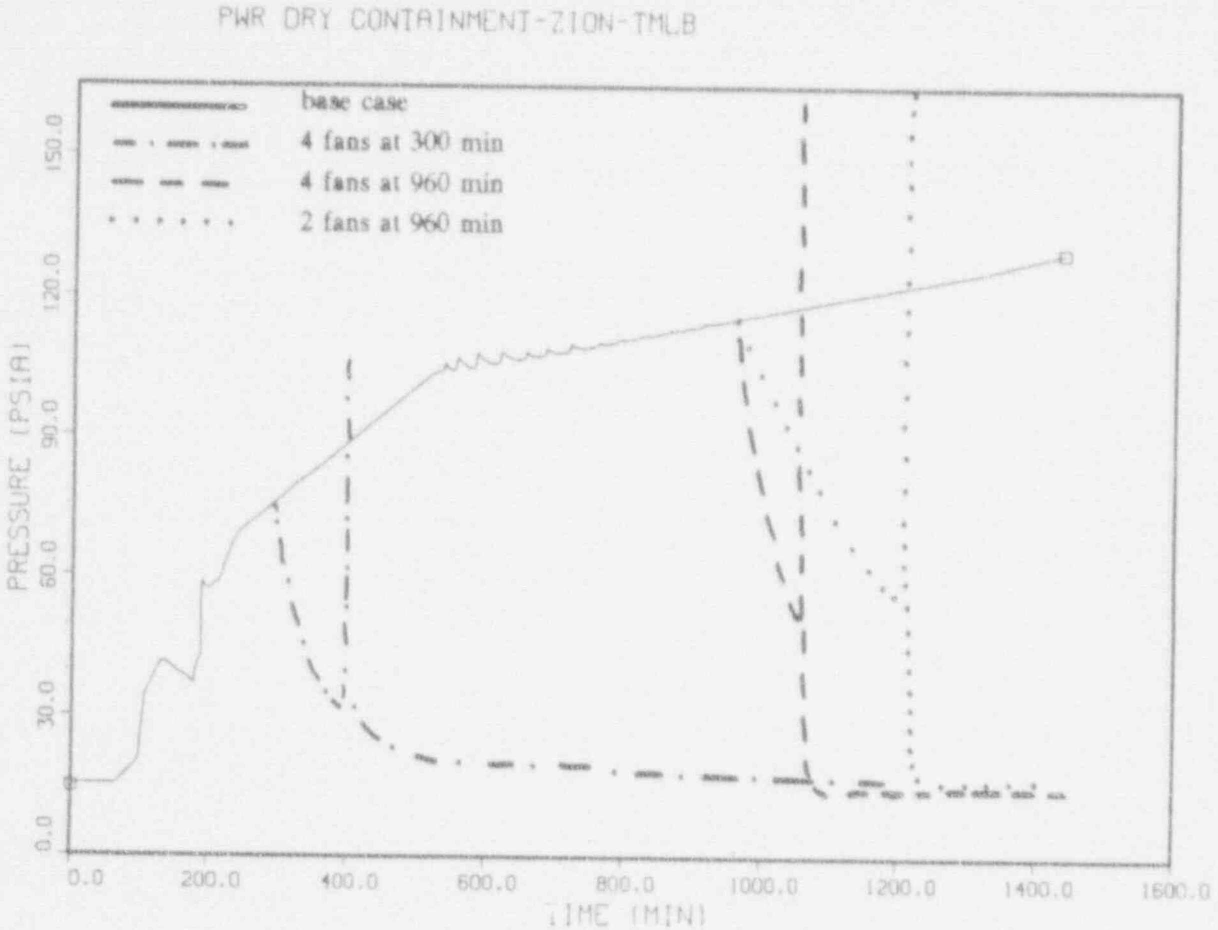


Figure 4.10 Effect of Containment Fan Cooler on Pressure for Zion SBO Sequence

gases. Currently, the ability to monitor combustibles or inertants locally is not available. Global hydrogen measurements can be made, but not CO or CO_2 measurements.

4.1.3.2 Surry plant

As discussed in Section 4.2.2, a potentially important late challenge to the Surry plant is basemat meltthrough rather than failure due to gradual overpressurization. Hence, mitigation strategies may focus on the flooding of the reactor cavity. However, the boil-off of the water introduced into the cavity will then make overpressurization a real possibility. Nevertheless, flooding may delay ultimate containment failure and provide additional time for recovery of CHR. Major concerns of this strategy are the time of flooding and mechanism of introducing water into the cavity. Because basemat meltthrough is a long-term challenge, flooding of the cavity should be implemented after reactor vessel breach to eliminate the potential adverse effects associated with early flooding.

The mechanism of adding water into the cavity depends on the cavity configuration. IDCOR [1] has classified the cavity configuration of all sub-atmospheric plants (Beaver Valley 1 & 2, Millstone 3, Surry 1 & 2, and North Anna 1 & 2) as a "type D" configuration (Figure 4.11). According to the IDCOR description, a type D cavity does not have good communication with the rest of the containment. NUREG-1150 [2] also describes the cavity configuration of Surry such that "water collecting on the floor (of the Surry containment) cannot flow into the reactor cavity." Therefore, modification of the cavity such as additional piping may be needed to provide water paths for flooding.

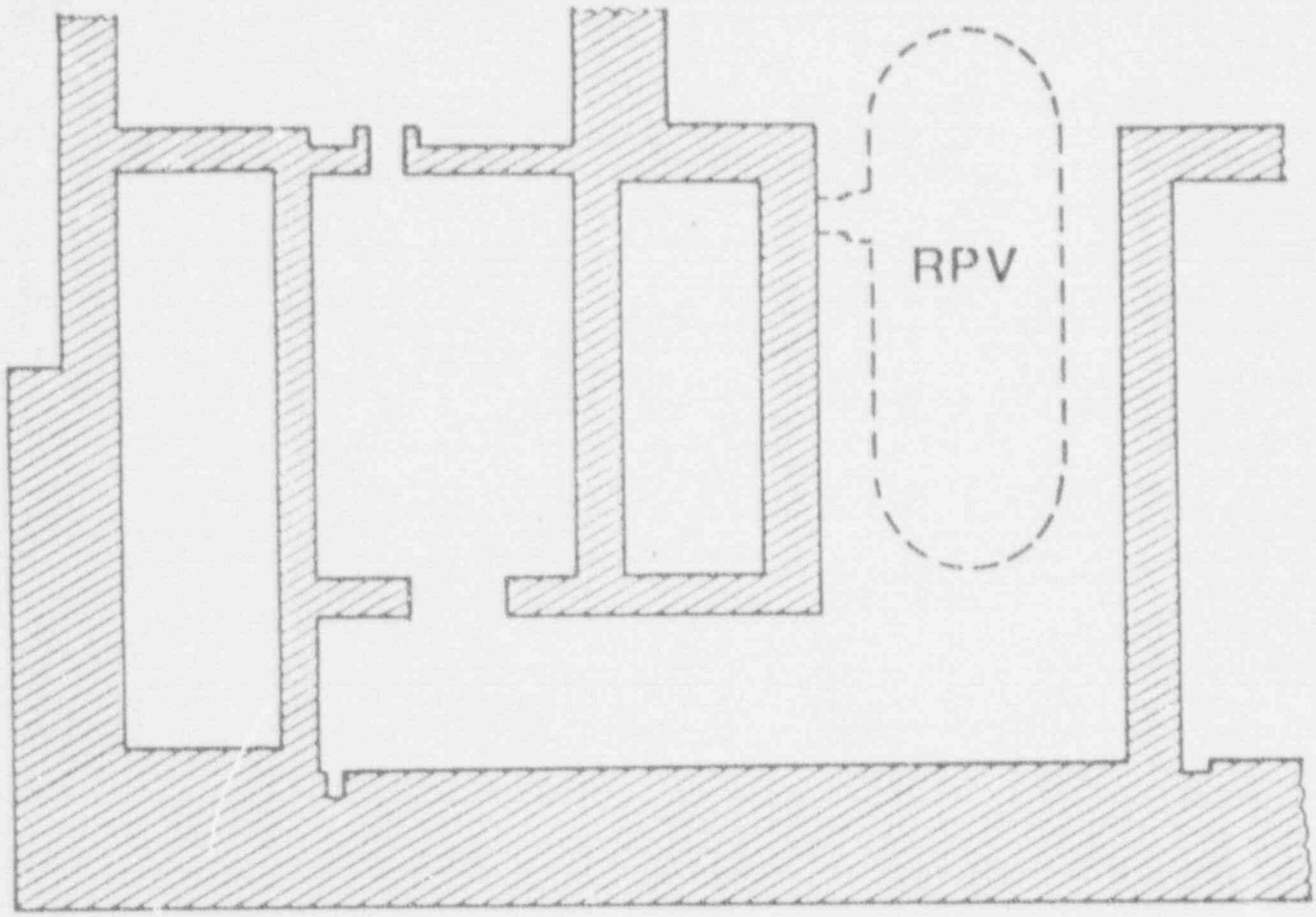


Figure 4.11 IDCOR Type D Lower Reactor Cavity Configuration

4.2 DCH Event

Direct containment heating (DCH) is the result of high-pressure melt ejection from the reactor vessel to the cavity at the time of vessel breach. Under high pressures, the debris could be dispersed into the upper containment. The fraction of debris which could be dispersed and involved with containment heating depends on a number of parameters, including the internal arrangement of the containment. If a large fraction of core debris could escape the reactor cavity, DCH would be a severe challenge to containment integrity. IDCOR classified PWRs according to reactor cavity configuration as shown in Table 4.2.

IDCOR's judgement of the potential for debris dispersal based on this classification has not been confirmed by experiment and involves a large uncertainty. Some existing experiments have cast doubt on IDCOR's estimate.

In a scoping experimental study of the extent of molten debris dispersal from PWR reactor cavities, Tutu, et al. [36], have conducted experiments using 1/42nd-scale model of reactor cavities of Zion, Surry, and Watts Bar. The experimental results show that the fraction of material dispersed from the reactor cavity are more than 0.99, 0.83, and 0.74 for the Zion, Surry, and Watts Bar plants, respectively. Tutu, et. al. [36], also developed dimensional analysis for the scoping study. A total of 10 dimensionless parameters are involved. Based on the scoping analysis, it is concluded that [illegible] debris would be ejected from the full-scale cavity of the Zion plant. For Surry and Watts Bar, at least 84% and 82% of the debris will be ejected out of the full-scale cavity, respectively. Note that the cavity configurations of the Surry and Watts Bar plants are classified as being of type D and C, respectively, by IDCOR (Table 4.2). These cavities are expected to retain a considerable amount or essentially all of the debris in the cavity. Thus, based on the experimental results, it appears that regardless of IDCOR's classification, a large fraction of core debris could be dispersed from these cavities.

Table 4.2 Lower Reactor Cavity Types of the IDCOR PWR Plants

LRC Type	Plant	Potential for Debris Dispersal During HPME
A	Braidwood 1 & 2 Byron 1 & 2 Zion 1 & 2	Debris would escape the reactor cavity
B	Indian Point 2 & 3 Seabrook 1 & 2 Trojan	Would not retain much debris
C	Catawba 1 & 2 Diablo Canyon 1 & 2 McGuire 1 & 2 Sequoyah 1 & 2 Vogtle 1 & 2 Watts Bar 1 & 2	Would retain a considerable amount of debris
D	Beaver Valley 1 & 2 Ginna Harris 1 Millstone 3 North Anna 1 & 2 Robinson 2 Surry 1 & 2	Would retain essentially all of the debris
E	South Texas 1 & 2	Little debris will escape
F	Arkansas 2 Calvert Cliffs 1 & 2 Millstone 2 Palisades	Most debris would escape
G	Oconee 1,2,3	Not much debris is expected to escape
H	Maine Yankee Palo Verde 1,2,3 WNP 3 Farley 1 & 2 Prairie Island 1 & 2 Summer 1 Turkey Point 3 & 4	Little debris is expected to escape
I	Point Beach 1 & 2 St. Lucie 1 & 2 Waterford 3	Not much debris is expected to escape
J	San Onofre 2 & 3	A considerable amount of debris is expected to escape
K	Arkansas 1 WNP 1	Not much debris is expected to escape
L	Bellefonte 1 & 2	Some debris would escape
M	Calloway 1 Comanche Peak 1 & 2 Wolf Creek Davis Besse 1	A good fraction of debris is expected to retain
N	Yankee Rowe	A reasonable fraction of debris is expected to escape

The impact of DCH strongly depends on the initial conditions of the reactor vessel and core debris. The containment pressure rise predicted by the CONTAIN code for two contrasting cases for the Zion plant are shown in Figure 4.12 [13]. The conditions of the two cases are given in Table 4.3. Case A5 assumes full pressure in the RCS prior to vessel breach, dispersal of 100% of the core melt into containment and fragmentation of the melt into fine particles. For this case, the peak pressure in the cavity region is about 3.2 Mpa, considerably in excess of the containment capacity (1.03 Mpa or 149 psia). Case C6 assumes partial depressurization of the RCS, limited dispersal

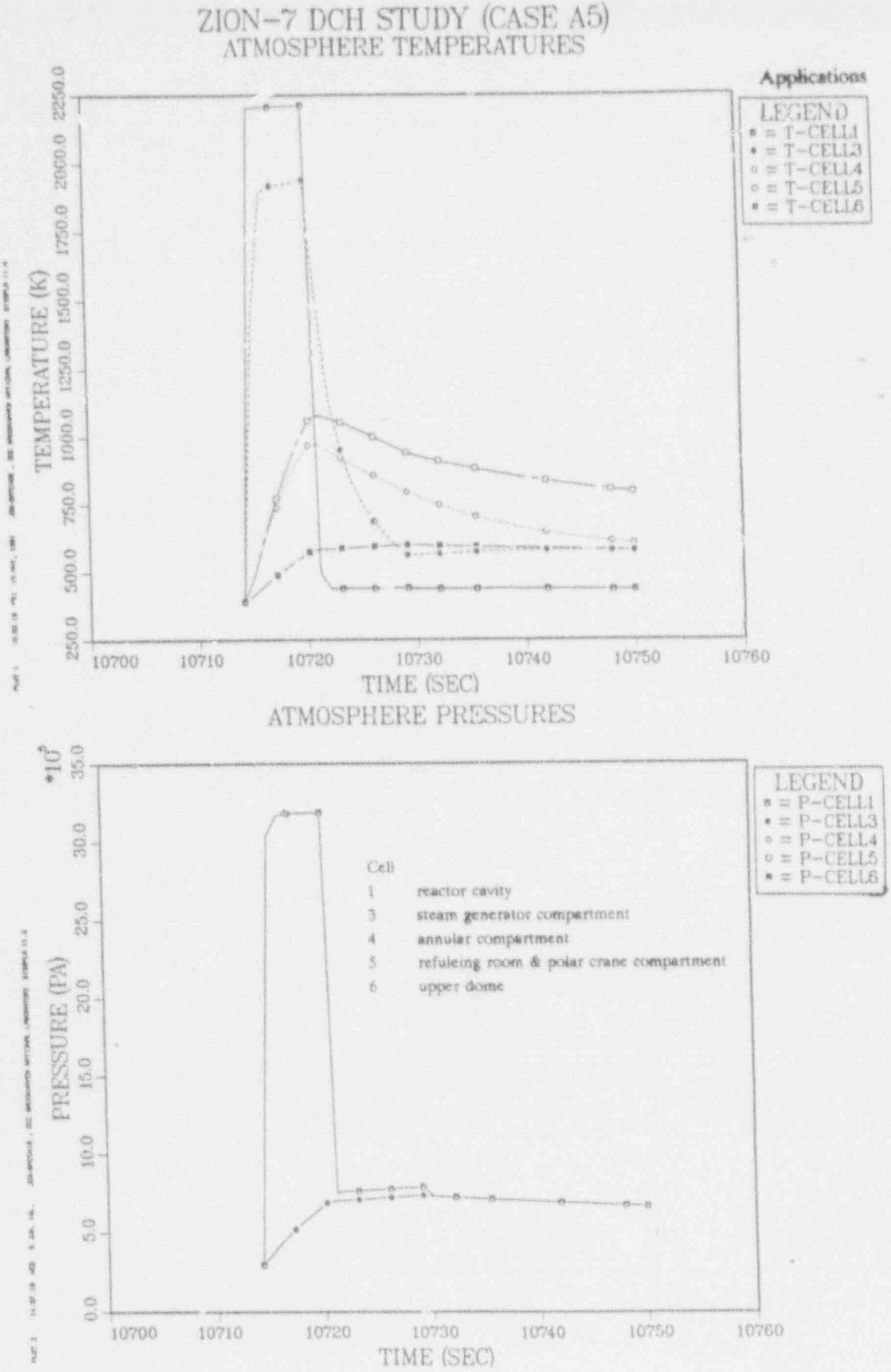
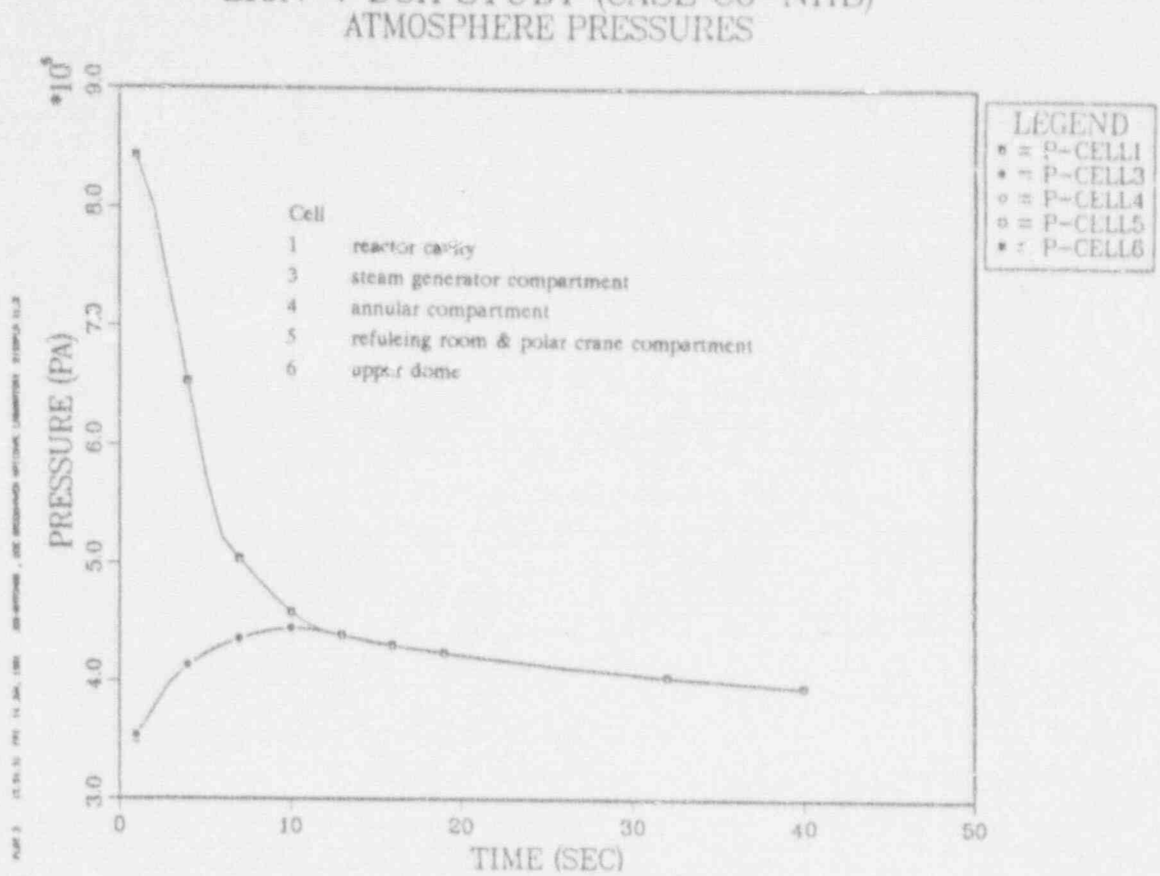


Figure 4.12 CONTAIN Predicted Containment Pressure for Zion DCH Event (Case A5)



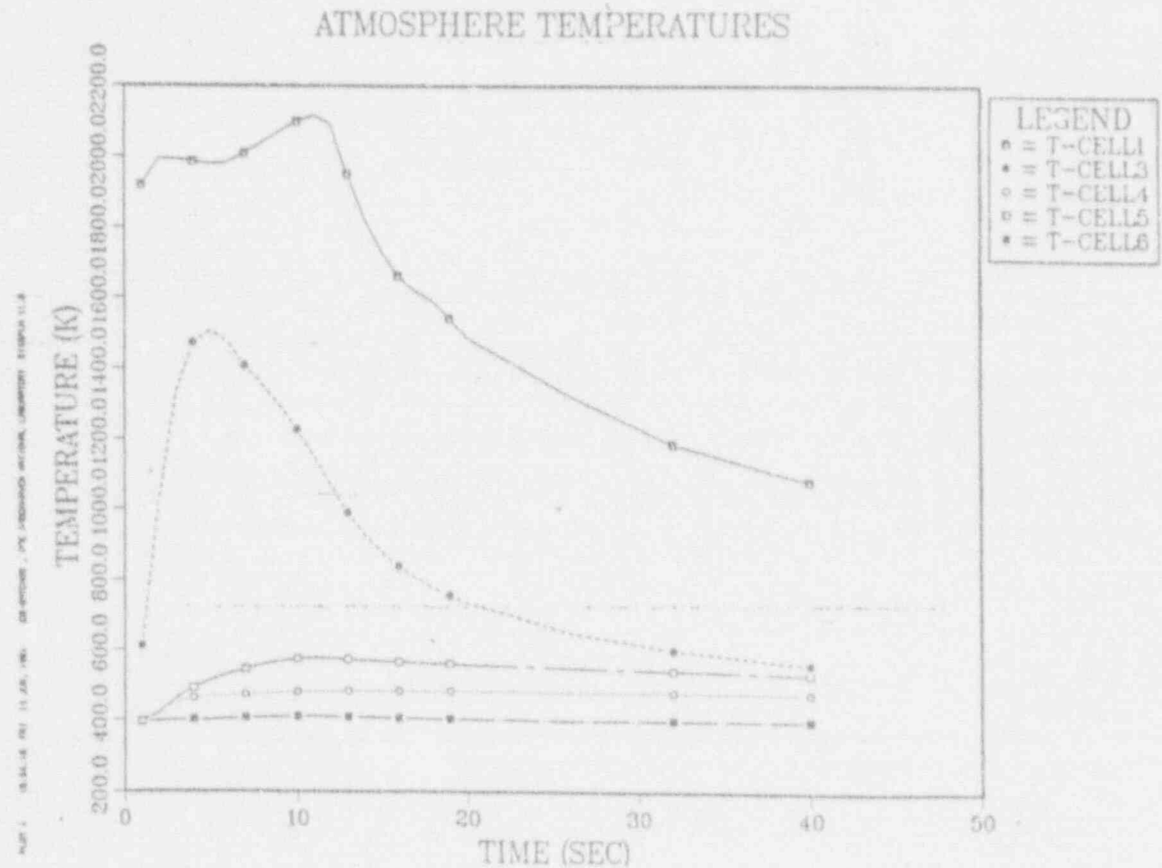


Figure 4.12 CONTAIN Predicted Containment Pressure for Zion DCH Event (Continued) (Case C6)

(30%) of core melt and relatively larger particle size. The predicted peak pressure for this case is much less than the estimated containment capacity. The comparison of the two cases illustrates the effect that assumed initial conditions have on the CONTAIN-DCH calculation.

Table 4.3 Initial and Boundary Conditions for Zion DCH Analysis

Case	A5	C6
Primary system pressure (Mpa)	16.4	3.4
Melt fraction	1.0	0.33
Particle size (mm)	0.5	1.0
Blowdown time (s) Debris Steam and H_2	 6 15	 12.2 12.2
Blowdown rate	Uniform	Linear
H_2 in containment (kg)	191	419
H_2 in vessel (kg)	321	14
Zr ejected (kg)	11000	3667
Fe ejected (kg)	18700	6233
React cavity	Dry	Dry

The mitigation strategy of RCS depressurization has been investigated in detail at the Idaho National Engineering Laboratory (INEL). Based on analyses for the Surry plant, INEL recommended depressurization after core uncovery. The strategy results were extended to other U.S. PWR plants [37]. Using Surry as the reference plant, five PWR subgroups were identified based on the normalized PORV capacity to RCS volume ratio. This ratio is considered as the overriding parameter affecting a plant's ability to depressurize. The higher this ratio, the more rapidly a PWR is able to depressurize relative to Surry. The five subgroups are:

1. Westinghouse PWRs with ratios greater than Surry's;
2. Westinghouse PWRs with ratios less than Surry's;
3. Combustion Engineering (CE) PWRs with PORVs;
4. Babcock and Wilcox (B&W) PWRs;
5. CE PWRs without PORVs;

A comparison of the normalized PORV capacity to RCS volume ratio is shown in Figure 4.13. The fifth group (CE PWRs without PORVs) was not included in the study. The conclusions of the INEL study are:

1. Intentional depressurization of the RCS using the late depressurization strategy has the potential to mitigate the effects of DCH in the Westinghouse Group 1 and 2 PWRs.
2. The Westinghouse Group 2 PWRs will probably experience substantially more core damage prior to accumulator injection than was predicted in the Surry analysis.
3. It is likely that the PWRs of the CE group will be able to use the late depressurization strategy to mitigate the effects of DCH.
4. The PWRs of the B&W group do not appear to have the capability to depressurize to a point where DCH is mitigated.

4.3 LOCA Sequences

A loss-of-coolant accident induced by coolant pump seal failure, a pipe break in the RCS or a stuck open PORV, should have the following signature: an increase of containment pressure and radiation level and a reduction of pressurizer pressure and water level. The operator will implement ERG E-O, which calls for verification of reactor trip, SI activation and auxiliary feed water (AFW) availability. ERG E-O directs the crew to E-1 to mitigate the LOCA based on high containment pressure, high containment radiation or high sump level. The CSS will be actuated manually or automatically, depending on the plant, after the containment pressure exceeds the actuation

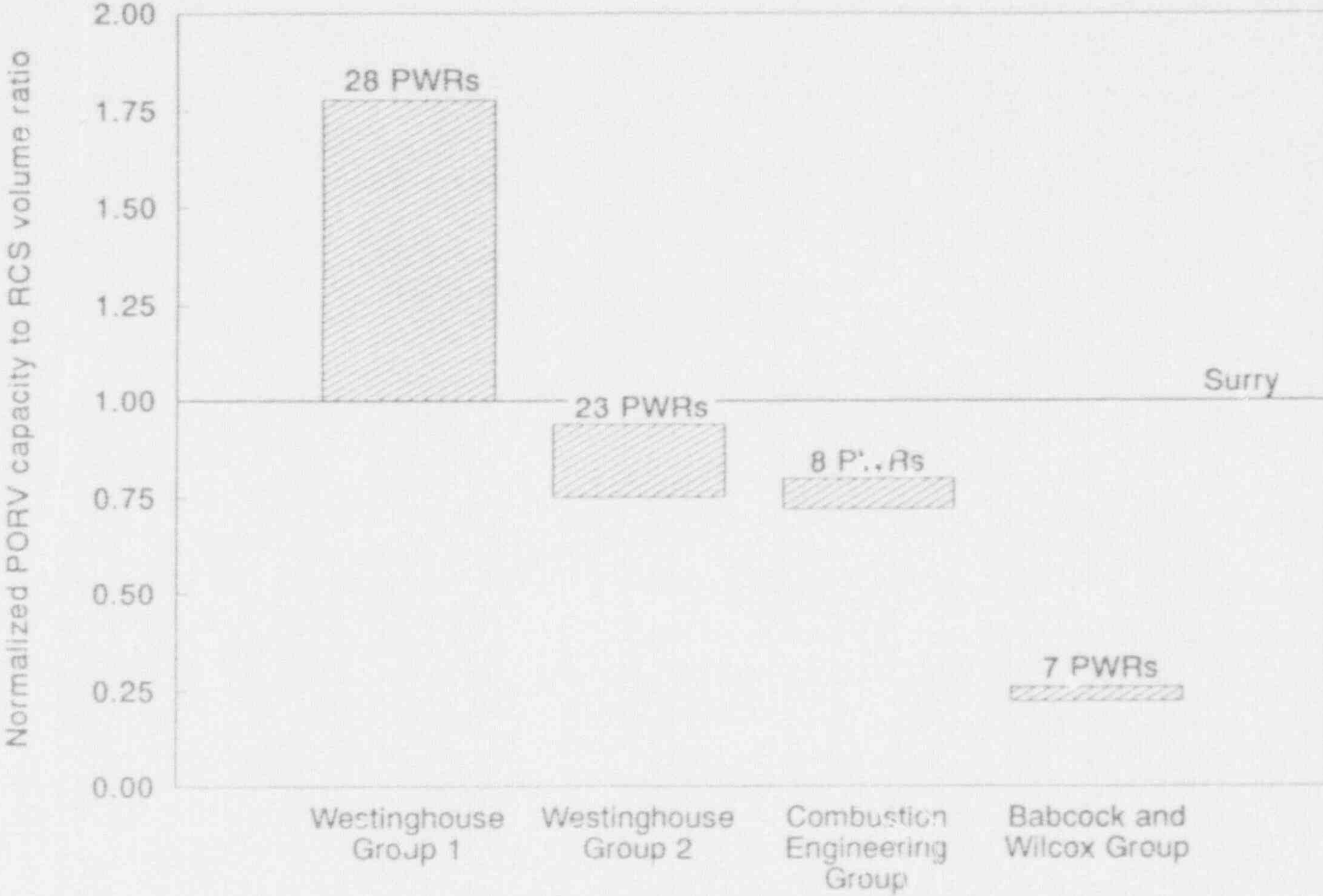


Figure 4.13 Grouping of PWRs for Late Depressurization Strategy Evaluation

pressure. The unavailability of the CHRS will be indicated in the control room in the form of the unavailability of the spray heat exchangers. The recognition of a LOCA in progress would call for the declaration of a site Area Emergency. If containment failure follows, or if containment venting is initiated, the conditions for declaring a General Emergency will be met. A General Emergency could be declared prior to containment failure if containment integrity is threatened. Typical actions required following the declaration of these emergencies are discussed in Section 4.1.

Zion and Surry are again selected as the representatives for the PWR atmospheric and sub-atmospheric plants, respectively, in discussions of containment challenges and mitigation strategies.

4.3.1 Zion

NUREG-1150 [2] has identified that reactor coolant pump seal LOCAs induced by the loss of component cooling water and service water are important accident sequences for the Zion plant. The event could lead to core damage if the service water/component cooling water system failed to recover in time to reestablish reactor coolant system inventory control. For the case in which the cooling system (component cooling water or service water) is recovered in time to provide injection from the RWST, core damage can still occur if the recirculation cooling fails.

Commonwealth Edison (the Zion licensee) has committed to perform the following actions with respect to the alternative water supply for component cooling [2]:

1. Auxiliary water supply is provided to each charging pump's oil cooler via the fire protection system. Hoses, fittings, and tools are locally available at each unit's charging pump area, allowing for immediate hookup to existing taps on the oil coolers, if required.

2. A formal procedure change was made to Abnormal Operating Procedure (AOP) 4.1, entitled "Loss of Component Cooling Water," providing instruction to the operators to align emergency cooling to the centrifugal charging pumps. Specific instructions are included for each charging pump with a diagram of the lube oil cooling valves and piping.

An STCP calculation for the pump seal LOCA sequence with an assumed failure size of 1.5 inches in diameter showed that core uncovery could occur at about 65 minutes [7]. Within this time-scale, the above emergency actions could be implemented. However, in the event of failure of these actions, the core could start to melt at about 94 minutes and the reactor vessel could fail at about 130 minutes. Because of the relatively small break size associated with pump seal failure, the RCS is likely to remain at high pressure and DCH could occur after vessel breach. To mitigate the potential DCH challenge, the RCS should be depressurized as discussed in Section 4.2.

In the event of vessel breach without DCH, STCP predicts the following containment response for the pump seal LOCA [7]:

1. The containment atmosphere is highly inerted due to the rapid water boil-off in the cavity region.

2. The cavity concrete erosion rate is relatively slow and would take about 36 hours for a complete penetration of the basemat.

3. The containment would fail at about 900 minutes as a result of overpressurization due to corium-concrete interaction, if the containment heat removal systems are not restored.

Thus, the containment challenge during the late phase of the accident is likely to be overpressurization. The mitigation strategies are similar to those discussed in Section 4.1, i.e., the restoration of containment sprays or fans or containment venting. Since sprays can effectively enhance aerosol removal, they should be considered as one of the CRM strategies.

Applications

As discussed in Section 3.2.3, caution must be exercised in the implementation of containment sprays because of the following two implications:

1. A PWR containment is generally inerted by steam under LOCA conditions. The restoration of containment sprays could de-inert the atmosphere and therefore, induce combustion. Although a global deflagration is not expected to raise the containment pressure beyond its capacity, a local detonation induced by flame acceleration (i.e., deflagration-detonation-transition, DDT) could severely threaten containment integrity. The potential for DDT depends on the containment subcompartment geometry and local gas concentrations.

2. For many PWR containments, the spray water is likely to be collected in the reactor cavity. There is a large uncertainty regarding the debris coolability by the overlying water pool. The industry developed MAAP code often predicts a complete quenching of the core debris if water is present in the cavity. Hence, a wet-cavity would eliminate the overpressurization challenge. On the other hand, the NRC sponsored CORCON code shows that an overlying water pool has little effect on debris coolability and, hence, can only mitigate, but not eliminate the overpressurization challenge.

A limited STCP analysis was performed for the S_2DC_rF sequence to qualitatively asses the spray strategy [7]. The S_2DC_rF sequence is a pump seal LOCA accompanied by failures of ECCS, containment spray recirculation, and fan coolers. In the analysis, three spray injection times were considered: prior to VB, immediately after VB, and late during the accident. Two cases with reduced spray rate and one case assuming the refill of the RWST late during the accident were included in the analysis. The analysis did not extend far enough to consider late steam condensation due to natural containment cooling. The results are:

1. The early initiation of containment sprays prior to the reactor vessel breach delayed the containment failure time by about six hours.

2. Initiation of sprays immediately after the reactor vessel breach induced a large hydrogen burn which did not threaten containment integrity. The containment failure time was delayed by about six hours.

3. The initiation of sprays late during the accident greatly reduced the containment pressure and delayed the failure time by more than 11 hours.

4. A reduced spray rate prolonged the spray time available using the RWST and this prolonged cooling further delayed the containment failure time by another one to two hours. The reduced spray rate also slows the deinerting effect of the atmosphere and, hence, reduces the potential for combustion.

5. Refilling the RWST for containment sprays during the late phase of the accident enhances the containment cooling effect and further delays the containment failure time by another three hours. However, the enhanced cooling caused a burn of 2100 pounds of hydrogen, which caused a pressure rise of 85 psia.

The above results are summarized in Table 4.4.

The containment challenge under LOCA conditions can be further demonstrated using the improved, multi-nodalization analyses from the MELCOR and MAAP codes. The accident sequence considered is a 2.5-inch break at the intermediate leg (S_2D). In both codes, the Zion RCS was represented by 14 nodes and the containment by 4 nodes. The MELCOR code predicted the failure of 4 penetration tubes (instrumentation tubes) in the reactor lower plenum between 190 to 250 minutes, and the failure of containment at about 27 hours. The MAAP code predicted that a single penetration tube failed at 226 minutes, and the containment at about 26 hours. Both codes indicated that the containment is unlikely to fail by either basemat melt-through or combustion. The major challenge to containment integrity under LOCA conditions is the slow overpressurization due to corium-concrete interaction. The

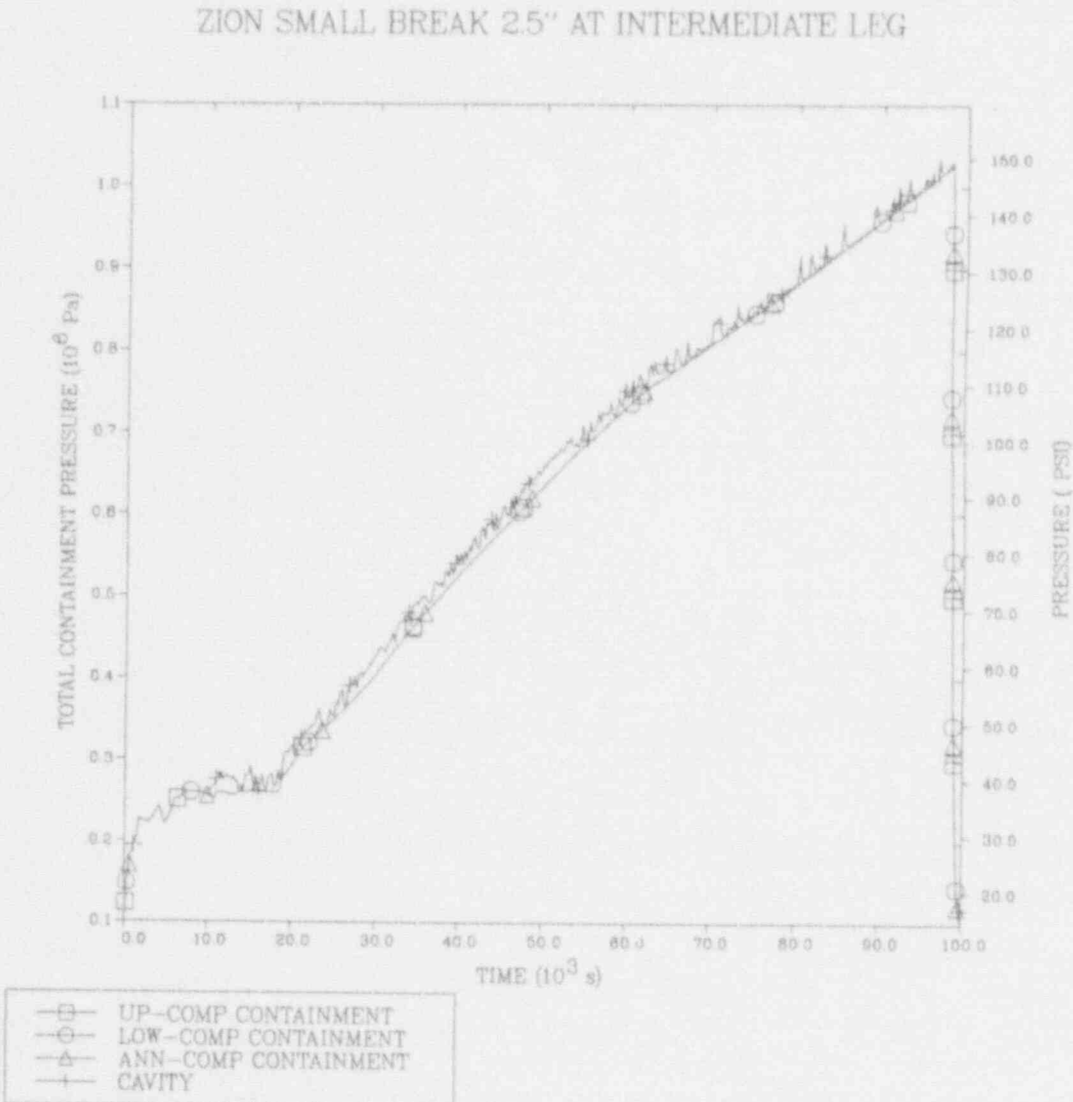


Figure 4.14 MELCOR Predicted Containment Pressure for the Zion S_2D Sequence

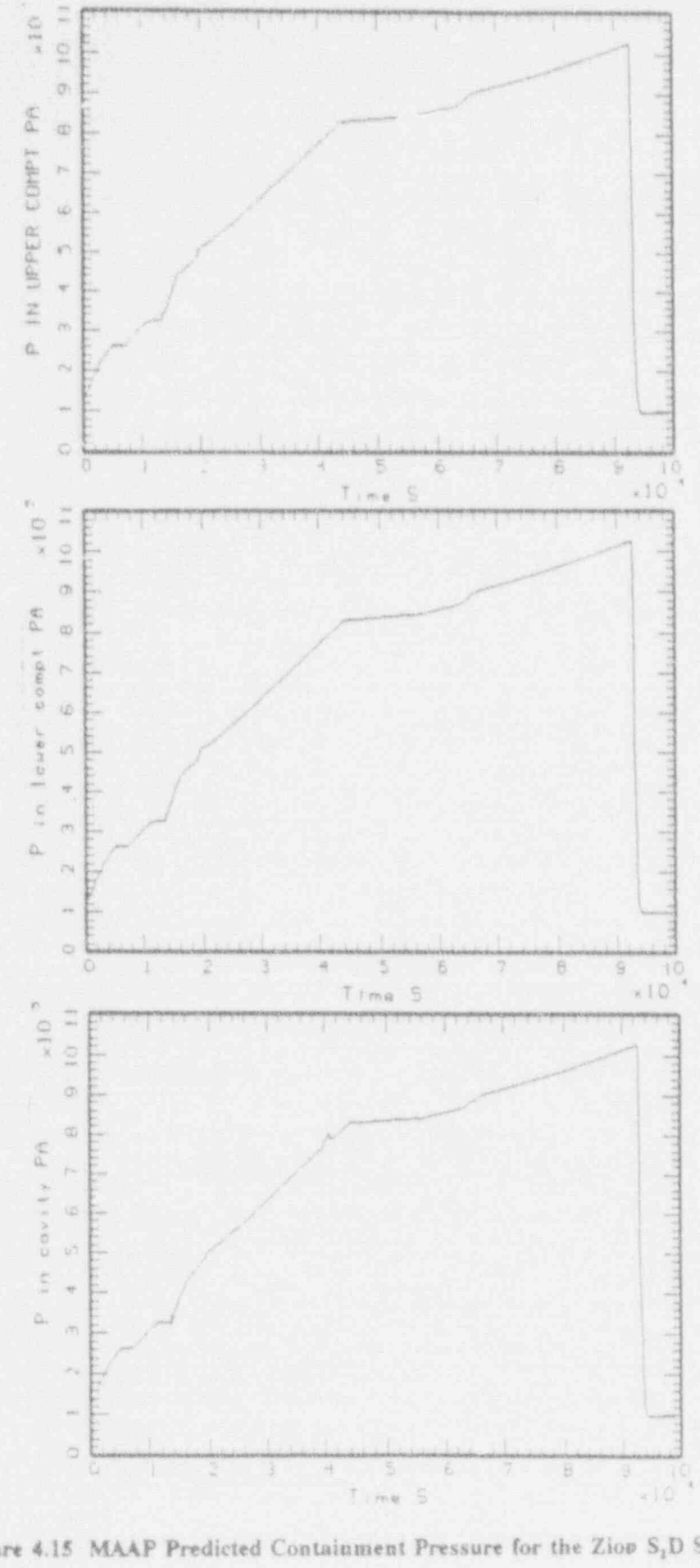


Figure 4.15 MAAP Predicted Containment Pressure for the Zion S_1D Sequence

Table 4.4 Summary of STCP Analysis for the Zion S_2DC_rF Sequence [7]

Case	Description	Containment Failure Time, min	H_2 Burn	Concrete Axial Erosion, cm
SF2	No operator action (no sprays)	900 (note 1)	No	106 (note 2)
SF25	Spray injection started prior to VB (31 min)	1260	No	84
SF45	Spray injection started at VB (134 min)	1248	Yes	84
SF8	Spray injection started at 800 min	1600	No (note 3)	85
SF580	Same as SF45, but 75% injection rate	1335	No (note 3)	84
SF390	Same as SF45, but 50% injection rate	1352	No (note 3)	86
SF8L	Same as SF8, but spray time doubled	1803	Yes	85

Notes
1. Containment failure at 149 psia due to overpressurization.
2. Cavity concrete erosion distance at end of 24 hours.
3. Although MARCH predicted no hydrogen burn, the gas concentrations are within the uncertainty of flammability. Combustion is possible.

MELCOR and MAAP predicted containment pressures are shown in Figures 4.14 and 4.15, respectively. Due to the slow rate of pressurization in the containment, venting could be implemented during the late phase of the accident, i.e., after a large quantity of aerosols has been deposited by natural settling processes.

4.3.2 Surry

NUREG-1150 [2] has identified that the combination of a small pipe break LOCA and the failure of coolant injection or recirculation is one of the major contributors to core damage for the Surry plant. The equivalent diameter of the break size could vary from 0.5 to 6 inches. All containment heat removal systems are available, but the continued heatup and boil-off of primary coolant leads to core uncovery. The time of core uncovery could vary from several minutes for a large break to several hours during a small break LOCA.

Analyses of the S_2D sequence for Surry were performed using STCP [7,34]. The results show that the containment could maintain its integrity through challenges such as hydrogen burning and pressurization. It is possible that the basemat will eventually be penetrated due to the attack of the concrete by core debris. For the case of a dry cavity, (i.e., containment sprays are not available during the small break LOCA event), the concrete rate is about 0.07 cm/min and a complete penetration of the basemat (about 3 m) would take about 72 h This late failure time would allow operators sufficient time to implement the cavity-flooding strategy.

4.4 SGTR Sequences

A steam generator tube rupture will expose the secondary side to full RCS pressure. These pressures are likely to cause relief valves to lift on the secondary side. While these valves are open, an open pathway from the vessel to the environment can result. Thus, the behavior of the secondary side safety valves is very important for the SGTR event. Based on considerations of plant emergency operating procedures, thermal-hydraulic calculations, and data from actual SGTR events, it was concluded in the NUREG-1150 study that there was a high likelihood that the safety valves would stick open.

The operating crew is initially lead to ERG E-O to initiate a reactor trip and to start the safety injection. Step 23 of E-O diagnoses a SGTR accident based on a high radiation level on the secondary side and directs the operator into Guideline E-3 "Steam Generator Tube Rupture." The E-3 Guideline provides necessary operator actions to terminate primary-to-secondary leakage and to prevent overfill of the steam generator. All the safety systems are assumed available at the initiation of E-3. The recommended steps include (1) checking the RCP trip criteria for further RCP operation, (2) identification and isolation of the broken SG, (3) control of the water level in the steam generators, including isolation of the broken steam generator from the AFW system, (4) initiation of RCS cooldown by dumping steam to the condenser from the intact SGs at the maximum rate using the PORVs or cooldown valves, (5) depressurization of the RCS in order to minimize break flow and to refill the pressurizer, (6) termination of safety injection to prevent repressurization after the termination of RCS depressurization, and (7) control of the RCS pressure and makeup flow to minimize break flow.

The above procedures when properly implemented will be able to control the rapid loss of reactor coolant inventory and restrain the release of radioactive materials to the atmosphere from the broken steam generator. Recently, a best-estimate analysis on system behavior during the SGTR transient was performed for the Kori Nuclear Unit (KNU) 3 & 4 [38]. The plant is a three-loop PWR designed by Westinghouse for a thermal power of 2785 MW (similar to Surry). The analysis used the RELAP/MOD1 code and focused on measures to control the break flow. Various cases were studied for transient boundary conditions and major operator actions based on the E-3 procedures. The study showed that the current E-3 procedure is adequate, but some recommendations were made for the improvement of the procedure. The recommendations are:

(1) Put the failed SG's PORV in the manual closed position with the isolation of the MSIV.

The lifting of the PORV of the failed SG at its set point pressure would result in an undesirable release of radioactive steam to the environment. Manual closure of the broken SG's PORV may limit radioactive releases. This action has little impact on the overall system behavior according to the analysis.

(2) Put the intact SG's MSIV in the manual open position before the start of RCS cooldown.

Because SG cooldown valves are located downstream of the MSIV, the isolation of the MSIV during RCS cooldown may disable further cooldown which, in turn, will repressurize the RCS and increase the leakage flow through the failed SG.

(3) Continue RCP operation based on RCS pressure for SGTR accidents that are not accompanied by an additional LOCA.

Unlike the situation during a small break LOCA, significant voiding of the reactor core is not expected during an SGTR accident. Thus, the operation of the RCP should be maintained to increase the RCS subcooling. The operator can monitor RCS pressure together with hot-leg subcooling to determine whether the RCP operation should be continued.

The EOP E-3 procedure assumes that all safety systems are available during the SGTR event. However, there are situations in which some safety systems may fail. For example, the emergency core cooling system could fail to switch to the recirculation mode because the containment is dry. Such sequences were analyzed for the Surry plant [39]. The results show that the reactor vessel could fail at 4 hours when both the injection and recirculation modes of the ECCS are not available. The reactor vessel is predicted to fail at about 15 hours when the failure of the recirculation mode was assumed. For the latter sequences, the mitigation strategy should focus on refilling the RWST with alternate water sources as discussed in Section 3.

5 Summary and Conclusions

Information on severe accidents, from NRC sponsored research efforts, as well as results from industry supported investigations, has been reviewed to identify the challenges a large dry containment of a PWR plant could face during the course of a severe accident, the mechanisms that cause these challenges, and the strategies that can be used to mitigate the effects of these challenges. The capabilities of existing plant systems and procedures that are relevant to containment and release management (CRM) have also been examined to determine their applicability to CRM and to determine possible areas for improvement. Important findings are summarized below.

5.1 Existing Accident Management Capabilities

Existing accident management capabilities are based on the NRC requirements described in NUREG-0737 [41] regarding emergency response, and NUREG-0654 [35] regarding radiological emergency plans and preparedness. The elements of these requirements that are most significant for CRM are the establishment of the technical support center (TSC), the availability of the emergency operating procedures (EOPs), and the requirements on plant instrumentation for accident monitoring.

In the accident scenarios examined in this report, the TSC will be activated and operational when CRM activities, beyond those of the current EOPs, are required. Since a wide variety of plant status conditions may occur and significant uncertainties on future accident progression exist, the availability of the TSC to take control of plant operations and to provide support to reactor operations is an important attribute for containment and release management in a severe accident.

The existing EOPs for a PWR with a large dry containment are based on the Emergency Response Guidelines (ERGs) prepared by the appropriate PWR vendor, i.e. Westinghouse, Combustion Engineering, or Babcock & Wilcox. The plant operations personnel can follow these procedures into a severe accident scenario. However, some of the assumptions on which the ERGs are based, are obtained from design basis accident conditions and may not be adequate for severe accident management after significant core degradation has developed. Modification of the existing ERGs regarding initiating and restricting conditions for accident response actions may be desirable to extend their applicability to accident phases beyond core damage.

The existing ERGs also concentrate on the restoration of core cooling and maintaining containment integrity under design basis loads. The mitigation of containment loading conditions that may occur after vessel breach or the mitigation of fission product release after containment failure are not emphasized. Additional guidelines for accident management after vessel breach or containment failure could therefore be beneficial.

A very significant potential problem with plant instrumentation for CRM is the lack of sufficient control room indications of containment variables during a station blackout (SBO) sequence. It may be desirable to provide an alternate power supply for some instrumentation that is important for making decisions on CRM activities. In addition, the identification of alternate methods to obtain containment variable indications in the absence of electric power will improve the availability of relevant information for CRM.

The survivability of plant instruments under severe accident conditions could also be a problem. The containment conditions, e.g. temperature, pressure, and radiation, that may occur in a severe accident may exceed the environmental conditions for which the equipment and instruments are qualified. Even though the equipment and instruments may survive under conditions well beyond their qualification conditions, their accuracy is not assured. A case by case analysis of the various types of instruments may be needed to determine their availability and reliability under severe accident conditions.

5.2 Interface Between Existing ERGs and CRM Strategies

An important goal of the USNRC's Severe Accident Management Program is to make innovative use of existing plant systems for accident management instead of resorting to costly hardware changes or additions. It is not surprising therefore, that many of the strategies described in the previous sections involve actions similar to ones called for in the current ERGs and often rely on the activation of systems designed to cope with design basis accidents. The CRM strategies differ from the existing ERGs primarily in terms of the conditions under which certain actions are undertaken and certain systems are activated. This includes operating systems in an anticipatory rather than a response mode, operating them beyond their design limits, as well as making use of non-safety grade systems in some instances. The boundary between current emergency procedures and those actions referred to as severe accident strategies is not a sharp one, and the interface between the two types of actions is complex.

The greater emphasis on anticipatory actions for CRM compared to current ERGs was illustrated by several strategies discussed in the previous sections. Such phenomena as HPME are too fast acting to allow remedial actions at the time of their occurrence, and therefore an advance action, like RPV depressurization, if high pressure RPV failure is deemed likely, may be advantageous.

Many actions called for in the ERGs, such as those dealing with secondary side heat removal, remain generally valid and useful in the severe accident regime as well. Others, however, may need to be modified or restricted: As discussed in previous sections, the activation of containment sprays during an advanced stage of a severe accident, where steam inerting may provide protection against a hydrogen burn, should be based on more than containment pressure and temperature levels. There are also a number of CRM strategies which have no direct counterpart in the ERGs. Cavity flooding is such a strategy, for instance.

How severe accident strategies in general, and CRM strategies in particular, are integrated into the plant emergency response will depend on many factors. Options include proceduralizing strategies so that they fit into existing EOPs, creating separate severe accident procedures, or providing more general guidance instead of specific procedures. There are advantages and disadvantages attached to all of these methods. While specific procedures lead to faster response than more general guidance, it is unlikely that all severe accident situations can be anticipated in sufficient detail to rule out the possibility that a requested procedure may be inappropriate for the particular situation. Some strategies may be easier to proceduralize than others. The resources of a particular utility can also determine the best method of CRM integration at a particular plant. If considerable expertise is available in the TSC to direct accident management, general guidance may be the optimum way to integrate CRM actions. On the other hand, if it is unlikely that a sufficient body of experts will be quickly available at the time of the accident, more specific advance direction should be developed in an accident management plan. In practice a combination of procedures and guidance is likely to be most effective in filling the needs of the operators, support staff, and accident management team.

5.3 CRM Strategies

Because of their large containment volume and high design pressure, PWR dry containments are relatively robust and provide considerable opportunities to maintain containment integrity and minimize the release of radiation following a severe accident.

Among the four hydrogen combustion modes which could potentially occur in such a containment, i.e. deflagration, diffusion flame, auto-ignition and detonation, the pressure loading and thermal effects of the three former modes are not expected to threaten the containment integrity. A local detonation caused by non-uniform gas distribution could challenge the containment integrity. The potential for local detonation depends on the containment construction, interior layout and other specific design parameters. PWR plants with subatmospheric design or steel shell design are more vulnerable to local detonation. However, for large dry containments, the risk from all combustion modes is deemed low enough that no modifications of these plants is necessary, although licensees should be cognizant of the potential for these events to occur. Plant specific combustion control should focus on promoting gas mixing and deliberate burning in order to keep the combustible gas concentration below the lean detonation limit.

The direct containment heating (DCH) associated with a high-pressure melt ejection (HPME) event appears to be an early threat to containment integrity. Many factors can influence the effects of melt ejection and some are not well enough understood to allow unequivocal statements regarding their influence on DCH. For instance, the impact of co-dispersal of water present in the reactor cavity during a HPME event involves considerable uncertainty. Flooding a cavity solely to mitigate DCH effects may not be justified based on present understanding. However, cavity flooding for some circumstances may be used to provide external cooling to the vessel, or in anticipation of cooling a debris bed. Therefore, the decision to flood has to take into account the range of possibilities in the particular accident under consideration.

However, mitigation or elimination of the DCH effect could be accomplished by RCS depressurization. INEL has developed a strategy based on late depressurization using PORVs, and has classified all PWRs into 5 groups for comparison of the effectiveness of the late depressurization strategy. The Babcock and Wilcox PWRs and Combustion Engineering PWRs with no PORVs are unable to depressurize the RCS by venting the primary system.

Overpressurization of PWR containments can occur during the late phase of an accident due to the buildup of steam and noncondensible gases. However, because of the large containment volume, for most PWR plants overpressurization is a slow process. In most cases, the ultimate capacity of the containment would be reached on the order of days. Under these circumstances, mitigation can be achieved by the restoration of containment cooling systems, using alternate water sources, or by a controlled venting. Restoration of containment cooling systems must be done cautiously so as not to de-inert the atmosphere and cause a sudden burn of a large quantity of combustible gases, which may have accumulated in the containment.

Basemat melt-through also is a potentially important challenge during the late phase of the accident for some containment designs. However, concrete erosion by the molten core debris is a very slow process and it would take days for the concrete to lose its structural integrity. The erosion of the concrete may be mitigated by flooding the reactor cavity. However, there is a large uncertainty regarding the effectiveness of cavity flooding since it depends on the cavity configuration and the state of the core debris in the cavity.

For some PWRs, containment bypass events provide a significant contribution to the risk estimates. The mitigation strategies that are discussed in this report include the isolation of the break line, reactor coolant systems (RCS) depressurization, refilling of the refueling water storage tank (RWST), flooding the break location, and the activation of auxiliary building fire sprays. These strategies are currently feasible for many PWRs. However, for some plants, modification of existing systems and/or procedures are required.

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