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Identification and Assessment of Containment and Release Management Strategies for a BWR Mark III Containment

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Abstract

This report identifies and assesses 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 BWR plant with a Mark III type of containment. Based on information available from probabilistic risk assessments and other existing severe accident research, and using simplified containment and release event trees, the report identifies the challenges a Mark III containment could face during the course of a severe accident, the mechanisms behind these challenges, and the strategies that could be used to mitigate the challenges. The strategies are linked to the general safety objectives which apply for containment and release management by means of a safety objective tree. The strategies were assessed by applying them to certain severe accident sequence categories deemed important for a Mark III containment because of one or more of the following characteristics: high probability of core damage, high consequences, lead to a number of challenges, and involve the failure of multiple systems.

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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 BWR plant with a Mark III type of 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. The Grand Gulf Nuclear Station is used as the example plant in this report, but some of the differences among the four domestic Mark III plants are also discussed.

The present report emphasizes the use of existing plant capabilities for severe accident management. The plant features that are important to containment and release management (CRM) of a BWR Mark III containment are reviewed to identify their function and performance under severe accident conditions. These include the plant systems, the resources needed to support their operation, the emergency response facilities, the emergency procedure guidelines (EPGs), and the instrumentation required to assess the plant and its environs during and following an accident. 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 BWR containments in general and Mark III plants in particular. Use was made of simplified containment and release event trees (CRETs) in both identification and assessment of strategies. One result of this examination is a safety objective tree which links the general safety objectives of containment and release management with the strategies identified as helpful in mitigating the challenges.

The strategies were assessed by application to certain accident sequences. The sequence categories selected for strategy assessment consisted of station blackout, ATWS, loss of containment heat removal, and containment bypass. These provide a range of accident characteristics which need to be considered: the initial condition of the reactor and the containment at the inception of the accident, the speed of accident progression, and the availability of major safety systems. The selected sequences also cover all the identified challenges and thereby allow all the strategies to be considered. Sequences with a significant probability of core damage or with the potential for high consequences are included in the assessment. The strategies discussed may, of course, also be of benefit in other sequences than the ones considered in this report.

Important CRM strategies are discussed in detail in this report to provide guidance for the development of symptom based strategies which could be considered for implementation. The most important points related to strategy implementation are discussed with emphasis on symptoms leading to strategy initiation, diagnostic concerns, downside risks, and operator action concerns. The challenges to which a Mark III containment is subjected during a severe accident are in many ways similar to those faced by the other BWR containments, i.e. Mark I and Mark II plants. Therefore many of the strategies are also similar. However an important additional challenge for Mark III's is burning of combustible gases. This challenge, and the strategies aimed at combustible gas control, are discussed at length in the report.

In addition to the BWR Emergency Procedure Guidelines the Grand Gulf Emergency Operating Procedures were used to estimate the operational response to a severe accident currently available at the plant. While the existing EPGs are designed primarily for plant conditions expected prior to significant core damage, CRM strategies consider plant conditions well beyond this point, including vessel breach and containment failure where release management becomes more important.

Although there are significant uncertainties in the understanding of some of the phenomena involved in a severe accident, the ability to predict accident progression accurately, and the plant capabilities under severe accident conditions, the strategies identified in this report were found to be in general effective based on their application during the accident sequences considered for the Grand Gulf Nuclear Station. Often a single strategy would have multiple beneficial effects on accident management (e.g., containment spray could reduce containment temperature and pressure, scrub fission products from the containment atmosphere, and provide water for corium

quenching). However some of the strategies may have significant adverse effects. In a Mark III containment the flooding of the reactor cavity, i.e. reactor pedestal area, may mitigate core-concrete interaction and reduce some loads associated with high pressure melt ejection, but may also lead to significant ex-vessel steam explosions. Current risk analyses indicate that the benefits of a flooded cavity still outweigh the drawbacks.

As for other containments, the lack of control room indications of containment variables in a Mark III could be a significant problem for accident management. This deficiency is particularly serious for a station blackout sequence, shown in NUREG-1150 to be a dominant severe accident sequence for a Mark III containment. The survival of plant instruments under severe accident conditions is also quite uncertain. The containment conditions, e.g., temperature, pressure, and radiation, that may occur in a severe accident may exceed the environmental conditions for which the instruments are qualified. These areas could benefit from additional research efforts.

An added suggestion based on the investigations performed for this report is that, during an actual accident, decision making for accident management may be enhanced through the use of simplified CRET's with updated plant status information and probability data to predict accident progression. When combined with a simple consequence prediction code and with the meteorological conditions and offsite activities already available, this could provide an integrated approach for accident progression and consequence prediction.

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The authors are grateful for Dr. Tim M. Lee's support during this project. Thanks are also due to the BNL staff who reviewed this report and provided valuable comments. Kara Roman should be commended for the typing and assembly of the manuscript.

List of Acronyms

ADS	Automatic Depressurization System
ANS	American Nuclear Society
APB	Accident Progression Bin
APET	Accident Progression Event Tree
ATWS	Anticipated Transient Without Scram
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CCI	Core-Concrete Interaction
CDF	Core Damage Frequency
CET	Containment Event Tree
CF	Containment Failure
CHR	Containment Heat Removal
CPI	Containment Performance Improvement
CR	Control Room
CRET	Containment and Release Event Tree
CRM	Containment and Release Management
CS	Containment Spray
CST	Condensate Storage Tank
CVS	Containment Venting System
DCH	Direct Containment Heating
DF	Decontamination Factor
DW	Drywell
ECCS	Emergency Core Cooling System
EOF	Emergency Operation Facility
EOP	Emergency Operating Procedures
EPG	Emergency Procedure Guidelines
EVSE	Ex-Vessel Steam Explosion
FCI	Fuel-Coolant Interaction
FP	Fission Product
FW	Fire Water
HCLL	Heat Capacity Level Limit
HCTL	Heat Capacity Temperature Limit
HEPA	High Efficiency Particulate Air
HIS	Hydrogen Ignition System
HPCI	High Pressure Core Injection
HPCS	High Pressure Core Spray
HPME	High Pressure Melt Ejection
HVAC	Heating Ventilating and Air Conditioning
IDCOR	Industry Degraded Core Rulemaking
IEEE	Institute of Electrical and Electronics Engineering
INEL	Idaho National Engineering Laboratory
ISL	Interfacing Systems LOCA
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Core Injection

LPCS	Low Pressure Core Spray
LTSB	Long Term Station Blackout
MAAP	Modular Accident Analysis Program
MCDF	Mean Core Damage Frequency
MEF	Mean Early Fatalities
MFD	Mean 50-Mile Dose
MLF	Mean Latent Fatalities
MOC	Mean Offsite Costs
MTD	Mean 1,000-Mile Dose
MSIV	Main Steam Isolation Valve
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
OSC	Operational Support Center
PCPL	Primary Containment Pressure Limit
PDS	Plant Damage State
PIV	Pressure Isolation Valve
PSP	Pressure Suppression Pressure
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Cooling System
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
SAM	Severe Accident Management
SARRP	Severe Accident Risk Reduction/Risk Rebaselining Program
SBO	Station Blackout
SC	Secondary Containment
SCSIP	Suppression Chamber Spray Initiation Pressure
SGTS	Standby Gas Treatment System
SOT	Safety Objective Tree
SORV	Stuck Open Relief Valve
SP	Suppression Pool
SPC	Suppression Pool Cooling
SPDS	Safety Parameter Display System
SPMU	Suppression Pool Makeup
SRV	Safety Relief Valve
SSW	Standby Service Water
STCP	Source Term Code Package
STSB	Short Term Station Blackout
TAF	Top of Active Fuel
TMCDF	Total Mean Core Damage Frequency
TPLL	Tail Pipe Level Limit
TSC	Technical Support Center
VB	Vessel Breach

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 procedures and guidance for severe accidents, that is, by planned management of severe accidents. While minor hardware modifications may in some cases be necessary to implement the resulting procedural changes or ad[illegible]uch 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.

Aspects of severe accident management have been considered in a number of previous NRC and contractor reports such as Reference 1. Brookhaven National Laboratory's (BNL's) contributions include NUREG/CR-4920, "Assessment of Severe Accident Prevention and Mitigation Features" [1], and NUREG/CR-5132, "Severe Accident Insights Report" [2]. In March 1990 NUREG/CR-5474, "Assessment of Candidate Accident Management Strategies" [3] was published by BNL. In this document a set of candidate accident management strategies, previously identified from various NRC and industry reports, such as NUREG-1150 [4], were assessed to provide information to individual licensees for consideration when performing their Individual Plant Examinations. The assessment focused on describing and explaining the strategies, considering their relationship to existing requirements and practices, as well as identifying possible associated adverse effects. The emphasis of the strategies assessed in NUREG/CR-5474 was on preventing or mitigating core damage, i.e., on arresting the accident progression in-vessel. The effects of the strategies considered were generally well understood and many of the strategies were found to be already implemented at some plants.

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 [5,6]. 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. While identification and assessment of advanced in-vessel strategies is being addressed by other NRC contractors, BNL is producing a series of reports dealing with the containment and release management part of a severe accident. The present report is one of this series. The mitigative strategies discussed here are often applied in situations where present understanding of the phenomena encountered is limited. Therefore, the uncertainty for these strategies is larger than for the strategies examined in NUREG/CR-5474. 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 written 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.

1.2 Objective and Scope

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 BWR plant with a Mark III type of 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 accident management plan.

The report can also furnish the reviewer of an accident management plan with a systematic overview of the challenges a Mark III 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 Mark III 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. Most, but not all, challenges can be met by available strategies.

1.3 Organization of the Report

The subsequent sections of the report are arranged as follows: Section 2 describes the approach taken for strategy identification as well as for strategy assessment. Section 3 describes the Mark III containment, the plant systems and resources, and existing severe accident management capabilities. A detailed examination of the containment challenges and the identification of the relevant containment and release strategies for a Mark III plant are presented in Section 4. At the end of Section 4 the challenges and strategies are systematically arranged in a "Safety Objective Tree." Section 5 presents the pertinent information for each of the strategies in a consolidated form. The application of the strategies during certain accident sequences is discussed in Section 6. Section 7 consists of a summary and conclusions. References are contained in Section 8.

2 Approach to Strategy Identification and Assessment

2.1 General Information

In order to optimize the effort of strategy identification and assessment maximum use was made of previously available information from Mark III related safety studies. The sources used most frequently are the NUREG-1150 study and the supporting reports related to Mark III [7-8, 9], the Mark III information produced for the NRC's SARRP [10-11] and CPI programs [12, 13], and the relevant IDCOR reports [14].

The BWR Owners' Group's Emergency Procedure Guidelines (EPGs), Revision 4 [15], were used as a baseline to gauge the guidance presently available to BWR Mark III licensees for their individual plants to respond to severe accident challenges. While it is recognized that the Emergency Operating Procedures of an individual plant may differ or go substantially beyond the EPGs, these guidelines provide the best available generic information on how Mark III plants will currently respond to a severe accident.

Using the existing BWR EPGs as a basis, additional operator actions in the form of accident management strategies were identified where appropriate and possible, and their anticipated effect on the accident was assessed. Included in the subsequent discussions is a description of the indicators that the operating staff would have (or would be lacking) at different stages of the accident to check the plant status, as well as those they would need to implement the suggested strategies.

For the suggested additional strategies, their impact on existing procedures, their effect on other aspects of accident management such as offsite emergency planning, and the human and material resources they would require, were also considered.

In the discussions which follow, when it is instructive to refer to specific plant features, Grand Gulf Nuclear Station is used as the example Mark III plant.

2.2 Strategy Identification Process

Numerous sources, referenced throughout this report, were consulted to obtain information on the challenges a Mark III containment could face during a severe accident, and the accident management strategies that can be used to prevent or mitigate these challenges. The challenges and strategies are identified in this report by a systematic examination of existing data, utilizing a simplified event tree structure, for accident progression. A description of the examination method and the outcome of this effort are presented in Section 4.

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 NUREG/CR-5474 [3] to organize the candidate strategies discussed there, and is similar as well to the safety objective tree structure used by INEL in NUREG/CR-5513, "Accident Management Information Needs," Volume 1 [16]. To achieve uniformity in terminology with other accident management reports where such a tree structure has or will be used, the terminology of NUREG/CR-5513 has been adopted here.

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 the Mark III containment is illustrated in Figure 4.6.

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. Nevertheless, no identification process can claim to account for all possible challenges and associated mechanisms, or to have identified all possible strategies.

2.3 Strategy Assessment Process

Previous history as well as the accident phase during which a challenge arises often play 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.

Since this report deals with containment and release related strategies, accident progression is tracked starting from a plant damage state. For the Mark III strategy assessment this tracking was accomplished through the use of simplified containment event trees whose top events consisted of events deemed important for accident management actions. These event trees have been used in the strategy identification described in Section 4, where some preliminary assessment of the strategies is also presented. A further assessment of the identified strategies, following the progression of selected accident sequences, is presented in Section 6.

To discuss strategy application it is convenient to distinguish among a number of phases during accident progression. These are: (1) the very early phase, before core damage has occurred, (2) the early phase, between the start of core damage phase to shortly after vessel breach, (3) the late phase, after vessel breach but prior to containment failure, and (4) a radiological release phase. These phases need not all occur in order. Depending on the accident, the radiological release phase can be entered from any of the other phases. Similarly, depending on the sequence and/or accident management actions, a recovery can be made from any of the first three phases. Figure 4.1 shows the relationship between the accident phases. It should also be noted that vessel breach is too sudden to allow for accident management actions during the actual time of vessel failure, but certain actions can be taken prior to failure with the purpose of mitigating the results of vessel breach. These actions are considered under the early phase.

3 Plant Capabilities and Severe Accident Management

The plant information that is important for containment and release management is discussed in this section. Section 3.1 describes the general features of the pressure suppression system of a Mark III containment, Section 3.2 discusses the plant safety and supporting systems that are important to severe accident management, and Section 3.3 describes existing accident management capabilities, particularly the BWR emergency procedure guidelines and the plant instrumentation required by NRC for plant condition assessment in an accident.

3.1 Mark III Containment System

The Mark III primary containment system is a pressure suppression containment system. It consists of (1) a drywell which encloses the reactor vessel, (2) an annular shaped pressure suppression chamber (wetwell) surrounding the drywell and containing a large volume of water (suppression pool), (3) a horizontal vent system connecting the drywell and the suppression pool, (4) containment isolation valves, (5) containment cooling systems, and (6) other service equipment. The primary containment system is designed to (1) condense the steam released during a postulated LOCA, (2) limit the release of fission products in an accident, and (3) provide a source of water for the emergency core cooling system (ECCS).

There are four commercial BWRs operating in the United States which utilize the Mark III containment design. These are Grand Gulf (1142 MWe), Clinton (930 MWe), Perry (1205 MWe), and River Bend (936 MWe).

The primary containment of Grand Gulf is a steel reinforced concrete structure, consisting of a vertical cylinder (3-6 feet thick wall), a hemispherical dome and a flat base (see Figure 3.1). This concrete provides both structural strength and biological shielding. A thin welded steel liner plate (0.25-0.5 inches thick) is used on the inside surface to form a leakage barrier. While the construction of the primary containment of Clinton is similar to that of Grand Gulf, Perry and River Bend use a free standing steel primary containment made with steel plate (~1.5 inches thick). The structure consists of a vertical cylinder with an ellipsoidal head and a flat bottom steel liner plate. The entire structure is anchored to the concrete basemat [13]. While providing both structural strength and a leakage barrier, the free standing steel primary containment offers little biological shielding. For Perry and River Bend this is provided by the reinforced concrete secondary containment described in section 3.1.3.

The main internal structures are, for the most part, common to all plants. The drywell is a cylindrical reinforced concrete structure with a flat roof slab. A circular opening in the roof slab is covered with a removable steel head to allow access for refueling. Enclosed within the drywell are the reactor vessel and a large portion of the reactor coolant pressure boundary including the coolant recirculation loops and associated pumps. This seismic Category I structure was designed to contain loss of coolant accident (LOCA) pressure transients and direct the resulting air-steam mixtures into the suppression pool. The drywell also provides support for the upper containment pool, as well as, radiation shielding, and protection for the containment from pipe whip, missiles, and jet impingement. The containment volume outside of the drywell consists of the upper dome and the lower wetwell (Figure 3.1). It includes the suppression pool, the annulus formed by the drywell outer wall and the primary containment inner wall, and the upper containment pool plus the volume above it [13].

Severe accident management, as defined in the NRC policy issue letter SECY-88-147 [5], includes the measures taken by the plant staff to (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) failing that, maintain containment integrity as long as possible, and finally (4) minimize the consequence of offsite release. Items (3) and (4) are the objectives of the present study. Containment characteristics and design bases relevant to these two objectives are discussed below for the primary containment, the secondary containment, and the suppression pool. In the following discussions Grand Gulf is used as the representative Mark III plant. It is important to note that there are variations among the four operating Mark III plants, and that while much of the subsequent discussion is generic to all plants, the variations may markedly affect the individual plant response to severe accidents. Some examples of these differences are discussed below.

3.1.1 Primary Containment

The primary containment's pressure capability and its failure mode under various containment loading conditions are important factors influencing the consequence of a severe accident. The ability of the primary containment to retain fission products, allowing natural deposition processes to occur, is another important factor affecting fission product release.

As long as the containment remains intact and is not bypassed the amount of fission products released will be insignificant. If the containment does fail, the consequence of fission product release will depend strongly on the timing and mode of failure. A larger failure size will result in a more rapid discharge, allowing less residence time for natural deposition, and consequently, in most cases a greater release of radioactive materials to the environment. Even with containment failure, if the drywell is not breached, the release to the environment will pass through the suppression pool and a large fraction of the fission products will be removed. A delayed containment failure will reduce the amount of radioactivity released by allowing more time for fission product decay, additional natural deposition in the containment, and a longer warning period for emergency response actions, i.e., evacuation, sheltering, and relocation. These issues are further discussed below.

3.1.1.1 Containment Pressure Capability and Failure Mode

The drywell of Grand Gulf is designed to withstand an internal pressure of 30 psid, an external pressure of 21 psid, and a temperature of 330 °F. The wetwell has an internal design pressure of 15 psig, an external design pressure of 3 psid, and a design temperature of 185 °F [13]. The primary containment leakage rate is limited to less than 0.35% free volume per day at design pressure and temperature [17]. For a comparison of containment design characteristics among the four operating Mark III plants see tables 3.1 and 3.2.

There is considerable uncertainty in estimating the ultimate containment strength and failure mode. In the Severe Accident Risk Reduction/Risk Rebaselining Program (SARRP) containment failure by overpressurization is assumed to occur at a pressure of 71 psia (with a break area of 7 square feet) [11]. In the NUREG-1150 study, probabilistic descriptions of containment failure pressure and failure mode were used. The mean failure pressure for Grand Gulf's containment is estimated at 55 psig [4]. For rapidly occurring pressure loads, the mean failure pressure of the drywell is estimated to be 85 psig [7].

During a severe accident, some actions, like containment venting, have to be based on extrapolated containment loading conditions and the expected containment performance under these conditions. Since such an action may result in unnecessary fission product release if implemented too quickly, i.e. before the containment's actual pressure limit is reached, a better knowledge of the containment's capability will increase the probability of making the right decision.

3.1.1.2 Containment Fission Product Retention

In the absence of additional sources, the amount of fission products in the containment atmosphere will decrease with time by natural deposition processes, and consequently, the amount of fission products released to the environment will be reduced if containment failure is sufficiently delayed. Additional time also allows more radioactive decay to occur before FPs are released. Containment fission product sources are twofold: those arising from the degradation of the core materials in the reactor pressure vessel (RPV), and those resulting from the attack of the concrete floor by the molten core debris after vessel breach. With STCP modelling most of the release from the vessel occurs before or at vessel breach. After vessel failure and the start of core concrete interaction (CCI) the CCI will reach a peak and then diminish to a negligible level within a few hours [18]. Although complete cooling of the debris may take a very long time, sufficient cooling to significantly reduce fission product release should take only a few hours. NUREG-1150 defines late containment failure, when fission products in the containment atmosphere have been greatly reduced by natural deposition processes, as 6 hours after vessel breach for the in-vessel release and 3.5 hours after the start of CCI for the ex-vessel release [19]. The models used in other severe accident codes may produce different CCI histories.

3.1.2 Suppression Pool

In a postulated LOCA the drywell is pressurized by the high energy coolant discharged from the primary system. This drywell pressure increase in turn forces the drywell atmosphere through the vent system into the suppression pool, where steam is condensed and noncondensible gases are released to the wetwell airspace. Vacuum breakers are provided between the drywell and the wetwell to relieve negative pressure events in which the wetwell pressure exceeds that of the drywell. For Grand Gulf there are three vacuum relief systems. The normal drywell vacuum relief system is provided to relieve a vacuum occurring due to normal variations in drywell conditions. This is not a safety system and on a LOCA signal these vacuum lines isolate. The drywell purge system provides vacuum relief after a LOCA through two lines drawing air from the containment into the drywell. The lines are set to open when the drywell pressure falls more than one psi below containment pressure. As a backup to the drywell purge system vacuum relief, the post-LOCA vacuum relief system is provided. It consists of two separate lines also drawing air from the containment into the drywell [13]. River Bend does not use vacuum breakers to provide protection against negative pressure events, but rather relies on reverse vent clearing. Reverse vent clearing occurs when a drop in drywell pressure relative to containment pressure causes the suppression pool water level to fall. Upon a large enough pressure differential the top row of vents uncovers allowing air to flow from the containment into the drywell [20].

The suppression pool also provides a heat sink for steam condensation during safety-relief valve (SRV) actuation. The SRVs are designed to control the primary system pressure. They are mounted on the main steam lines inside the drywell with the relief lines discharging into the suppression pool.

The suppression pool is an alternate water source for the high pressure core injection systems (RCIC and HPCS), and the sole water source for the low pressure ECCS systems (LPCS and LPCI) and the containment spray (CS) systems [8]. LPCI and CS are different operating modes of the RHR system and, as such, share components of the RHR system.

The energy deposited into the suppression pool during an accident can be removed from the suppression pool via the RHR heat exchangers. The ultimate heat sink for the RHR heat exchangers is provided by the plant standby service water (SSW) system. The suppression pool plays a very important role in fission product removal during a severe accident. It provides significant fission product scrubbing of any flows passing through it. Since it is the water source of many safety systems, pool conditions, such as water temperature and water level, affect the performance of the engineered safety features of these systems. A brief discussion of the role of the suppression pool in severe accident management is presented below.

3.1.2.1 Suppression Pool Decontamination Factors

Suppression pool scrubbing is particularly effective for fission products produced in-vessel and released through the SRV spargers. The decontamination factor (DF) used in the NUREG-1150 analysis for in-vessel releases ranges from 1.1 to 4000 with a median value of 60 [4]. In comparison, the DF for ex-vessel releases and flows passing through the horizontal vents is smaller. The DF values used in the NUREG-1150 analysis range from 1 to 90, with a median value of 7.

After the RPV is breached, fission products are released to the drywell. Part of these releases will pass through the vent system and be scrubbed by the suppression pool. Drywell failure would allow fission products to pass directly into the wetwell airspace without the benefit of suppression pool scrubbing. As demonstrated by the large DF range given above, there is considerable uncertainty in the effectiveness of suppression pool fission product scrubbing capability. Nevertheless, the integrated decontamination factor is in general significant and it is important to assure that any release to the environment should pass through the suppression pool, if possible.

3.1.2.2 Suppression Pool Temperature

The suppression pool temperature is one of the control variables in the BWR emergency procedure guidelines (EPGs) and is monitored and controlled under both normal and accident conditions [15]. Reactor vessel depressurization is required if pool temperature exceeds the heat capacity temperature limit (HCTL) to avoid exceeding either the suppression chamber design temperature, or the primary containment pressure limit (PCPL).

Suppression pool temperature is controlled by the operation of the RHR heat exchangers, which are designed, with redundancy, to remove the reactor decay heat in a design basis accident. However, excessive pool heat up may occur in some accident sequences. The pool temperature will increase if the heat removal rate of the RHR heat exchangers is not sufficient to handle the heat influx, as can happen in an ATWS event, or if the containment cooling function of the RHR system fails, as happens in a TW sequence (Loss of long term containment heat removal).

Loss of suppression pool temperature control may result in exceeding the design temperature and pressure limits. High pool temperature may cause a resuspension of the FPs in the SP, and a flashed SP will add to the driving force causing the release of the containment atmosphere, and the fission products it contains, to the environment. A high suppression pool temperature will also increase the potential for late iodine release from the suppression pool, which is another source term issue addressed by expert elicitation in NUREG-1150.

3.1.2.3 Suppression Pool Water Level

The suppression pool water level is another EPG control variable [15]. The suppression pool loses its pressure suppression capability if its water level is too low. However, there are also problems associated with a high water level. A high water level can result if water sources other than the suppression pool are used for either core injection or containment spray. A high water level raises concerns about (1) the loads associated with clearing the water slug initially in the SRV line during SRV discharge, and (2) flooding the vacuum breakers between the drywell and wetwell.

Following the guidance of the BWR EPGs, the Grand Gulf emergency operating procedures provide specific directions and procedures to control suppression pool water level in an accident. The suppression pool makeup (SPMU) system is provided in most Mark III plants to supply water to the suppression pool following a LOCA. This system allows the upper containment pool to be gravity fed through two lines into the suppression pool. Each line contains two motor-operated valves in series. Emergency procedures are also provided to add water to the suppression pool with the RCIC or the HPCS systems if the SPMU system is unavailable. Procedures are available as well to remove water from the suppression pool through test lines connected to the condensate storage tank (CST), if the water level is high [21]. Since the suppression pool water could be highly contaminated in a severe accident, finding means to remove excessive suppression pool water for safe storage in a leaktight tank is important.

3.1.3 Secondary Containment

Enclosing the primary containment is the secondary containment. The performance objective of the secondary containment is to provide a volume, completely surrounding the primary containment, which can be used to hold up and dilute fission products that might otherwise leak to the environment following a design basis accident. The standby gas treatment system (SGTS) is designed to provide a mixing of the secondary containment volume and maintain the volume at a slightly negative pressure. The exhaust air required to maintain the negative pressure is discharged through SGTS filters.

The secondary containment for Grand Gulf is made up of the auxiliary building, a reinforced-concrete structure which surrounds the lower portion of the primary containment, along with the enclosure building covering the upper part of the primary containment. The Grand Gulf enclosure building is a steel-framed structure covered with metal siding which is attached to the auxiliary building roof. During normal operation the secondary containment airspace is maintained at a slight negative pressure by the fuel handling area and auxiliary building ventilation systems [22]. The secondary containment for Clinton has a similar design to that of Grand Gulf.

The secondary containment used in the Perry and River Bend plants consists of a reinforced concrete structure surrounding the primary containment, a fuel building, and the auxiliary building. The concrete structure, also called the shield building, provides biological shielding for outside structures, as well as protection from severe meteorological conditions and externally generated missiles. It consists of the flat foundation mat, a cylindrical wall and a shallow dome. The annulus between the shield building and the steel primary containment is filled with reinforced concrete to an elevation of about 24 feet above the top surface of the basemat. The region between the secondary and primary containments is maintained at a slight negative pressure during normal operation [23,20].

The secondary containment characteristics and systems that can affect the release of fission products to the environment are discussed below.

3.1.3.1 The Standby Gas Treatment System

The standby gas treatment system (SGTS) is designed to limit the environmental release of radioisotopes, which may be released from either the containment or fuel handling area, under accident conditions [8]. It provides a filtered and controlled release of the secondary containment atmosphere. The height of the release point varies from about 31 to 200 ft above ground for the Mark III plants [20]. Although the SGTS was not designed for severe accidents and may not have the capacity to handle the releases from a particular severe accident, a judicious use of the system may mitigate fission product release. A brief description of the SGTS for Grand Gulf is presented below.

The SGTS of Grand Gulf is located in the auxiliary building. Operation of the SGTS automatically starts in response to the following signals: high drywell pressure, low reactor vessel water level, or high radiation level in either the fuel handling area ventilation exhaust or the fuel pool sweep system exhaust. The SGTS consists of two full capacity enclosure building recirculation fans (each with a flow capacity of 17,000 cfm), two full capacity exhaust fans (each with a maximum flow of 4300 cfm), two full capacity charcoal filter trains, and two redundant sets of the associated ductwork, dampers, and controls [8]. Each charcoal filter train consists of the following components in sequential order: demister, electric heater, prefilter, HEPA filter, charcoal adsorber, and HEPA filter fan [22]. Iodine is removed by activated charcoal beds, which have a typical design adsorption efficiency of 99 percent for elemental iodine and organic iodide. The presence of adsorbed water on the charcoal surface will substantially affect this efficiency by reducing the surface area available for the trapping of volatile forms of radioactive iodine.

During a severe accident, a significant amount of aerosols may reach the secondary containment. These can plug the HEPA filters and reduce the flowrate through the charcoal filter trains. Eventually filter elements may tear due to excessive aerosol plugging. Given a failure of the HEPA filters of the SGTS a significant amount of charcoal bed adsorption may still be maintained [24]. Even in the case when both HEPA and charcoal filters fail, operation of the SGTS may still be desirable because of the paths and release point associated with the SGTS. On the negative side, the operation of the SGTS blowers may reduce the residence time of fission products in the secondary containment and, in the event of a loss of system filters, accelerate the fission product release.

3.2 Plant Systems and Resources

The plant systems and resources that can be used for severe accident management include those that are designed for emergency containment cooling under accident conditions and those that, through innovative application, can be used to perform accident management functions they were not originally designed for. NUREG/CR-5474 [3] has discussed in detail some accident management strategies related to innovative use of systems and resource management. Although the emphasis of NUREG/CR-5474 is on maintaining core cooling, the strategies concerned with locating and managing additional water, power, and pneumatic supply resources are equally applicable to containment and release management (CRM). The plant systems and resources that are important to CRM are discussed briefly below. Grand Gulf plant parameters are used for illustration.

3.2.1 Primary Containment Cooling and Water Supply

In Grand Gulf the containment cooling system (a fan-cooler system) is used during normal plant operation to maintain the primary containment atmosphere at design conditions outside the drywell of 80 °F and 60% relative humidity. The drywell cooling system (also a fan-cooler system) is used during normal operation to maintain average drywell design conditions of 135 °F and 50% relative humidity. The residual heat removal (RHR) system is used during an accident for emergency cooling of the primary containment. The RHR system uses the suppression pool as its water source and can be operated in suppression pool cooling (SPC) mode, or containment spray mode for containment cooling. A short description of these systems for Grand Gulf is given in this section. Also discussed are the RHR system's alternate water sources which can be used in case its normal water source, the suppression pool, is not available.

The containment cooling system in Grand Gulf consists of three 50% capacity recirculation cooler units distributed inside the containment. Each cooler unit consists of a 50-hp fan and a cooling coil. The capacity of each unit is 27,500 cfm. Cooling water to the coil is supplied by the plant chilled- water system.

The drywell cooling system consists of six fan-cooler units distributed throughout the drywell. Each unit consists of two full-capacity 11.5-hp fans and two full-capacity cooling coils. Each unit has a capacity of 12,000 cfm. During normal operation cooling water to the coils is supplied by the plant service water. In a loss of offsite power event, the standby service water system is used to supply the coils. Normally, only one fan and one coil from each unit operate with the second fan and coil on standby. These fans, as well as the unit coolers, trip automatically in the event of an accident, but can be manually restarted from the control room for use during an accident.

The RHR System in Grand Gulf has three trains, each with its own motor-driven pump, piping, valves, instrumentation and controls. Two of the three RHR trains can be used to accomplish the safety function of containment heat removal through operation of either the SPC mode or the containment spray mode. Each train has a pump rated at 7450 gpm. Both of these trains have two heat exchangers in series with a combined heat removal capacity of approximately 54 MW per train [17]. The tube sides of the RHR heat exchangers are fed by the standby service water system. The RHR pumps take suction from the suppression pool and are powered by the emergency diesel generators if offsite power is not available. The SPC mode of the RHR system is manually actuated and controlled. Since the SPC mode shares common systems with the RHR core injection and containment spray modes, its use is prohibited if a core injection or a spray signal is generated subsequent to SPC mode initiation and the RHR system automatically realigns to either core injection or containment spray mode [8]. The containment spray mode is started manually or initiated automatically on high containment pressure with a 10 minute delay.

The RHR system is designed to take suction from the suppression pool. An alternate water source is needed in a severe accident if this normal water source is not available either due to an alignment problem or because the suppression pool water temperature is high enough to raise concern about insufficient net positive suction head (NPSH) and possible damage to the pumps. Alternate water supplies can be obtained from crossties with other plant systems or from sources outside the plant.

In Grand Gulf a crosstie with the standby service water (SSW) system is already available. The SSW system, which contains the plant's ultimate heat sink, takes suction from the mechanical cooling tower basins, which only supply the SSW system and have no other function. A crosstie can also be made with the fire water (FW) system. The fire water system in Grand Gulf has two diesel-driven pumps as backups to an electric motor driven pump, and, therefore, could supply water to the primary containment during station blackout when ac power from both offsite sources and the standby diesel generators is not available. All pumps can provide a flow capacity of 1500 gpm at 125 psig, and are capable of taking suction from either of the two 300,000 gallon fire water storage tanks [8]. A cross connection which allows the fire water system to use the condensate and refueling water storage and transfer system as a backup fire water source is also provided [22].

The use of alternate water sources not presently available to the RHR system has also been suggested in previous investigations [3]. Crossties may be arranged to make these water sources available to the RHR system. For plants that have multiple units, crossties of similar systems from different units exist in many cases. These include the cross-connection of the water storage tanks of various water supply systems. Water sources from outside the plant include the municipal water system via the use of portable pumps, or the use of offsite tanker trucks or railroad tank cars.

3.2.2 Electric Power and Pneumatic Supply

Electric power and pneumatic supplies are required to support the operation of safety equipment. Their availability is critical to plant safety and accident management. A brief discussion of the electric power and pneumatic supply systems for Grand Gulf is given in the following along with their availability and the possible additional sources and backup systems that can be used in a severe accident.

The Grand Gulf station has two independent sources of offsite power, one from three 500 kV overhead transmission lines and the other from a 115 kV overhead line. The standby ac power is supplied by three onsite diesel generators, which start automatically on a total loss of offsite power. One of the diesel generators is dedicated to HPCS system, while the remaining two supply power to the other essential loads. Each diesel engine and its related generator circuit breaker are tripped by protective devices under abnormal conditions such as high coolant or lube oil temperature, low coolant or lube oil temperature, or low fuel oil pressure [22].

There are three independent Class 1E 125 V dc systems for Grand Gulf. Each system is comprised of a 125-V battery bank, two battery chargers, a load center, and a distribution panel. There are also six non-Class 1E 125 V dc systems, two of which are connected in series to supply 250 V dc power to auxiliary loads. The dc power can provide control and switching power to safeguard systems and apparatus, dc auxiliaries, and motor-operated valves during station blackout (loss of all ac power). In a station blackout (SBO) event at Grand Gulf, the dc power is expected to last at least 12 hours with appropriate load shedding measures.

Strategies to extend the availability of electric power have been discussed in NUREG/CR-5474 [3]. For example, the availability of ac power, from either offsite or emergency diesel generators, can be enhanced by crossties with other units in a multiple unit station; the operation of the diesel generators can be extended by bypassing certain protective trips or changing their trip setpoints if such action will not result in early diesel generator failure; and battery life can be extended by shedding non-essential loads or with the use of portable battery chargers. Detailed discussions of these strategies related to loss of power can be found in NUREG/CR-5474.

In Grand Gulf the pneumatic supplies are provided by the instrument air and service air systems. The instrument air system for Unit 1 consists of one full-capacity compressor, complete with filter, air dryer, and aftercooler. The instrument air system of the uncompleted Unit 2 is also present and operational. It is connected to Unit 1 through available crossties and can be used as a backup supply. These systems deliver compressed air at 110 psig to support the operation of safety related equipment. The service air system for Unit 1 consists of one full-capacity compressor. It is arranged as an automatic backup supply to the instrument air system through the use of a control valve which opens on reduced pressure in the instrument air line [8]. Vital components, such as MSIVs and SRVs, are provided with accumulators to assure reliable function without compressor operation. Some plants also utilize a long-term, backup, safety-related, pneumatic supply to the ADS valve accumulators.

NUREG/CR-5474 [3] has discussed strategies to enable emergency replenishment of the pneumatic supply for safety related air operated components. The options for additional air supplies include: diesel air compressors and additional onsite storage of bottled gas systems.

3.2.3 Containment Spray System

The containment spray (CS) system is designed to keep the pressure and temperature loads on the primary containment within their design basis limits. The CS system is an operating mode of the RHR system and shares components with other operating modes. Two of the three RHR loops can be utilized by the CS system. Each of these loops forms a completely independent and redundant CS train containing its own motor-operated valves, motor-driven pump, heat exchangers, and spray headers. The CS system normally takes suction from the suppression pool and delivers a flow rate of 5650 gpm. The capability to use the SSW system as a CS water source is also available via an existing crosstie (see Section 3.2.1). There are three CS headers in each train located at different elevations in the upper section of the wetwell. The Mark III design does not have a drywell spray system.

In addition to its design function of containment pressure and temperature control, the CS system is also a significant severe accident management tool because of its ability to remove fission products from the containment atmosphere. If given sufficient time, containment sprays are very effective in reducing airborne concentrations of fission product aerosols and vapors. This can greatly reduce releases in those scenarios involving failure of both the containment and the drywell [4]. Iodine removal rates for Grand Gulf containment sprays are calculated to be 6.7 hr^{-1} for elemental iodine and 1.66 hr^{-1} for particulate iodine [22].

There are possible adverse effects associated with the operation of the CS system. These include unacceptable containment negative pressure loads caused by spray operation and containment deinerting due to steam condensation allowing the possibility of subsequent hydrogen combustion. The impact of these potential adverse effects on containment integrity and the subsequent release profile should be assessed before spray decisions are made. More discussion of these items can be found in later sections of this report dealing with the BWR EPGs and the loading conditions during severe accidents.

3.2.4 Primary Containment Venting

Containment venting has been recognized as an important accident management strategy and has been incorporated in the BWR EPGs. It is used to prevent containment failure by providing a controlled release of the containment atmosphere if the containment pressure approaches a specified limit.

The venting system of Grand Gulf is described below. The important issues for a successful implementation of a venting strategy, the adverse effects of venting, and possible improvements for successful venting are also discussed.

3.2.4.1 Containment Venting for Grand Gulf

The containment venting system (CVS) can be used to prevent the primary containment pressure limit from being exceeded given failure of suppression pool cooling and containment sprays [8]. The vent path used in the CVS is the 20-inch purge exhaust line of the containment ventilation and filtration system. There are four air-operated dampers in the line which are normally closed and fail closed on loss of air. Two of the dampers close on a containment isolation signal, while the other two close on SGTS initiation. The initiation of containment venting requires that operators override the isolation relays and reopen each of these dampers. The CVS discharge is on the roof of the auxiliary building [8].

3.2.4.2 Important Issues for Containment Venting

Currently, the only objective of containment venting is to prevent overpressure failure. To reduce the probability of unnecessary radioactivity release, the venting pressure should be set at the highest possible value without failing the containment. However, there are other considerations for determining venting pressure. In the BWR EPGs, the primary containment pressure limit (PCPL), i.e., the pressure for venting initiation, is defined to be the lesser of either (1) the pressure capability of the containment, (2) the maximum containment pressure at which vent valves can be opened and closed to reject decay heat from the containment, (3) the maximum containment pressure at which SRVs can be opened and will remain opened, and (4) the maximum containment pressure at which vent valves can be opened and closed to vent the RPV. In certain severe accidents, when containment pressure rises rapidly, venting may have to be initiated before the pressure limit is reached to avoid containment failure.

As discussed in Section 3.1.1.1, there is significant uncertainty in the pressure capability of the containment. Although a Mark III containment is tested to 1.15 times the design pressure (typically 17.25 psig) and the use of this pressure assures containment structural integrity, a higher PCPL is desirable as well as practical because it is plausible that containment pressure capability is much higher than the design pressure. Since containment strength may deteriorate as containment temperature increases, containment venting decisions may need to account for temperature also.

The pressure rise in the containment during an accident is directly related to the energy input to the containment atmosphere and provides an indication on the venting area required. However, the loss of energy absorption capability of the suppression pool due to venting needs to be considered also. A saturated suppression pool can still absorb additional energy as the containment is pressurized because the saturation temperature increases with containment pressure. Once the containment is vented containment pressure will either stay constant or decrease, and the suppression pool will stop absorbing additional energy, or possibly even release energy to the containment atmosphere through pool flashing.

A successful implementation of a containment venting strategy requires knowledge of the potential adverse effects associated with containment venting. This will help to identify ways to avoid or minimize these effects. Possible adverse effects of containment venting include loss of plant safety equipment due to containment depressurization and suppression pool flashing and fission product releases to the environment.

The ability to operate the vent paths required for a successful containment venting depends on (1) the availability of electric power and pneumatic supplies, (2) the ability to defeat isolation signals and perform valving and lineup operations in the secondary containment, and (3) the time and manpower available to perform the required venting operations. Most of the above requirements are dictated by the accident sequence that is occurring. Thorough investigation of vent path operability under various severe accident conditions, to identify problems and methods to surmount these problems, and clearly defined guidelines or procedures are essential for the success of containment venting.

3.3 Existing Accident Management Capabilities

Accident management capabilities currently existing in nuclear power plants are based on NRC requirements described in NUREG-0737 regarding emergency response capability [25] and NUREG-0654 regarding radiological emergency response plans and preparedness [26]. The facilities and procedures established in response to these requirements will be used during a severe accident for accident management. The effectiveness of these capabilities in severe accident management needs to be evaluated and information obtained from this evaluation can be used to modify or extend existing capabilities to improve their effectiveness.

The elements of the existing capabilities that are most important to the investigation of CRM include (1) emergency response facilities, (2) existing emergency operating procedures (EOPs), and (3) the plant instrumentation and safety parameter display system (SPDS). These items will be discussed below. General ideas on extending existing emergency procedures for severe accident management and the relationship between the extended and existing procedures are also discussed.

3.3.1 Emergency Response Facilities

The emergency response facilities include (1) the technical support center (TSC), (2) the operational support center (OSC), and (3) the emergency operations facility (EOF). These facilities are designed to support the control room (CR) during an accident, and will be activated according to the severity of the emergency. Four emergency classes (in order of increasing severity) are defined by NUREG-0654 [26]. They are (1) Notification of Unusual Event, (2) Alert, (3) Site Area Emergency, and (4) General Emergency.

The TSC is an onsite facility located close to the control room (within 2-minute walking time) and is designed to provide management and technical support to the personnel located in the control room during emergency conditions. Its activation is optional for the Notification of Unusual Event emergency class, but is required for Alert and higher classes. Upon activation of the TSC, designated personnel shall report directly to the TSC, and the TSC shall achieve full functional operation within about 30 minutes. The EOF is an offsite support facility for the management of overall licensee emergency response. This involves coordination of radiological and environmental assessment, and determination of recommended public protective actions. Its activation is optional for Notification of Unusual Event and Alert emergency classes but required for Site Emergency and General Emergency classes. The OSC is an onsite facility where predesignated operations support personnel can assemble during an accident. While the OSC is not specifically required by NRC regulations, both the TSC and EOF are required facilities.

When activated, the EOF is primarily responsible for the management of corporate emergency response resources and radiological emergency response plans. The TSC is responsible for the management of plant operations and provides technical support to reactor operations, thus taking the primary responsibility for the containment and release management (CRM) of interest to this report. Nevertheless, the EOF assumes overall responsibility for accident management upon its activation.

As noted above, the TSC is activated during the Alert emergency class. The Alert emergency class is defined in Reference 33 as follows: "Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the

EPA Protective Action Guideline exposure levels." Examples of initiating conditions for the Alert emergency class include: Loss of offsite power and loss of all onsite ac power; failure of the reactor protection system to initiate and complete a scram which brings the reactor subcritical, and primary coolant leak rate greater than 50 gpm. The plant conditions when CRM is required will most likely exceed these conditions, and therefore the TSC is expected to take control of plant operations and emergency response functions and make accident management decisions until the EOF is activated.

The TSC staff consists of technical, engineering, and senior designated licensee officials. The TSC personnel are provided with reliable data to determine site and regional status. They determine changes in the status, forecast the status and take appropriate actions. They are also provided with accurate, complete, and current plant records essential for the evaluation of the plant under accident conditions. However, additional guidelines and calculational aids prepared specifically for severe accident management may be useful in the TSC for more effective management.

3.3.2 Existing Emergency Procedure Guidelines (EPGs)

The emergency operating procedures (EOPs) are plant procedures that direct operator actions needed to mitigate the consequences of transients and accidents that have caused plant parameters to exceed reactor protection system set points or engineered safety feature set points, or other established limits [27]. The technical basis of an individual plant's EOPs are the BWR Emergency Procedure Guidelines (EPGs), Revision 4, prepared by the General Electric Company [15].

The BWR EPGs Revision 4 are functionally divided into four guidelines: (1) the RPV control guideline, (2) the primary containment control guideline, (3) the secondary containment control guideline, and (4) the radioactivity control guideline. Three of the four guidelines, i.e., Guidelines 2,3 and 4, are related to containment and release controls. The EPGs are symptomatic guidelines: Operators' actions are based on the values of the control variables, e.g. suppression pool temperature, and not on their judgement regarding what types of events are occurring.

Because the procedures are symptom based, the operator should be able to follow the procedures well into a severe accident by observing selected plant variables. However, some of the assumptions on which the EPGs are based may not be adequate for severe accidents. Operator actions limited to the present EPGs may not be optimum or even appropriate for severe accident management. Additional guidelines for severe accidents may need to be developed, and the decision to switch from one guideline to another during the progression of a severe accident may also need to be addressed. The EPGs that are related to containment and release control are briefly discussed in the following sections.

3.3.2.1 Primary Containment Control Guideline

The purpose of the primary containment control guideline is to maintain primary containment integrity and protect equipment in the primary containment. The entry conditions to this guideline used in the Grand Gulf EOPs are (1) suppression pool temperature above 95 °F, (2) drywell temperature above 135°F, (3) drywell pressure above 1.23 psig, (4) suppression pool water level outside the range of 18.34' to 18.81', or (5) containment temperature above 90°F [21]. The entry conditions given above are symptomatic of both emergencies and events which may degrade into emergencies. Entry into the procedures does not necessarily mean that an emergency has occurred.

The primary containment control is concerned with monitoring and controlling of the temperature and pressure of the drywell, the temperature and water level of the suppression pool, and the temperature of the containment.

According to the guidelines, the operator should first try to control the variables within predetermined limits using normal plant equipment. If this fails and containment conditions further degrade, the operator should then carry out the RPV control guideline to shutdown the reactor, to perform emergency RPV depressurization, and/or to take additional actions to secure containment integrity and equipment protection, actions such as containment venting and spraying or switching the suction source for emergency cooling system pumps.

The design assessment loading conditions, such as those from LOCA and SRV actuation, are the basis for some of the actions specified in the BWR EPGs. Both the suppression pool air bubble load from SRV actuation and the pool swell load from LOCA vent clearing depend on the amount of noncondensible gases discharged to the suppression pool. The containment loads from these events after significant core degradation has occurred will be different than those used for design assessment. Consequently, containment damage may happen prior to the time expected in the EPGs if the loads under severe accident conditions are more serious. Since in a severe accident the SRV loading condition occurs only if the RPV is not depressurized, and the LOCA loading condition occurs only for a high pressure vessel breach, both loading conditions could be avoided by keeping the RPV depressurized.

3.3.2.2 Secondary Containment Control Guideline

The purposes of the secondary containment control guideline are to maintain the integrity of the secondary containment, to protect the equipment in the secondary containment, and to limit radioactive releases to the secondary containment and the environment. The secondary containment control guideline is concerned with monitoring and controlling the temperature, radiation levels, and water levels in the secondary containment. In general, when the value of any of the above control variables exceeds its predefined maximum operating limit the operator is instructed to take actions to maintain the value within the limit and, if this fails, to isolate the systems that are discharging into the problem area. Finally, if the conditions further deteriorate, the operator should take action by entering the RPV control guideline to shutdown the plant or to carry out emergency RPV depressurization.

3.3.2.3 Radioactivity Release Control Guideline

The purpose of the radioactivity release control guideline is to limit radioactive release outside the primary and secondary containments. Similar to the secondary containment control guideline, the approach taken in the EOPs is to direct actions to determine and isolate the source of the release and at the same time to ensure that the operators take proper action with respect to plant operation even if the source cannot be readily identified or if isolation efforts are not successful. For reasons similar to those discussed above, during a severe accident the plant may have deteriorated to a state such that the procedures provided in this guideline become impractical. Operator efforts should then be concentrated on reducing offsite radioactivity release using plant features discussed in Sections 3.1 and 3.2, such as (1) reducing the amount of fission products in the primary containment atmosphere, (2) providing fission product scrubbing in the primary containment by containment spray or pool scrubbing, and (3) enhancing the fission product retention capability of the secondary containment.

3.3.2.4 Additional Guidelines for Containment and Release Management

The existing EPGs extend well beyond the design basis accidents and include many actions appropriate for severe accident management. However, the existing EPGs may not be appropriate or effective for the management of a severe accident after significant core damage has developed for the following reasons: (1) The initiating and limiting conditions for some operator actions are derived from assumption of containment noncondensible gas content that may not be appropriate for severe accidents after core damage, (2) some of the procedures that cover the early stage of an emergency are not applicable in the late stage of an accident but may still command the operator's attention and thus become a distraction, and (3) if a severe accident progresses to a certain stage, the emphasis shifts to the control of fission product release which is not specifically covered in the existing EPGs.

To focus the attention of the operating personnel on severe accident management a separate guideline specifically prepared for severe accident management, instead of modifying and extending existing EPGs to cover the whole range of severe accident conditions, may be desirable. Some of the later parts of the existing EPGs may be incorporated into the severe accident management (SAM) guideline for a smoother transition. A logical transition point from existing EPGs to SAM guidelines is when significant core damage has occurred. SAM includes both in-vessel and ex-vessel management, the present study of containment and release management (CRM) considers only the ex-vessel part of SAM.

The CRM guidelines may have a similar general structure as that of the existing EPGs, by specifying operator actions based on plant symptoms, to guard against serious misdiagnosis. However, the CRM guidelines should be more flexible because of the large uncertainties in our understanding of plant capabilities and severe accident phenomenologies. The guidelines should pay adequate attention to (1) innovative use of available equipment and resources for accident management, and (2) directing actions to recover lost, or identify alternate, equipment and resources. As discussed in Section 3.3.1 the TSC is most likely activated and in control of plant emergency functions when CRM activities are demanded. The TSC has the capability to assess severe accident conditions and is suitable to manage the accident following more flexible guidelines. However, specific TSC personnel should be designated to take definite responsibilities to assure successful severe accident management.

3.3.3 Instrumentation, SPDS, and Environmental Qualification

The instrumentation required to assess the plant and its environs during and following an accident is described in Regulatory Guide 1.97 (Rev. 3) [28]. There are five types of variables to be monitored during an accident and according to their importance to safety they are separated into three design and qualification criteria categories.

Certain control room instrument indications that are essential to the emergency response capability of the nuclear plant are displayed on the Safety Parameter Display panel. NRC requirements for the Safety Parameter Display System (SPDS) design are specified in NUREG-0737 [25]. The SPDS is required to provide a concise display of critical plant variables to the control room operators to aid them in rapidly and reliably determining the state of the plant. It shall provide sufficient information to plant operators about (1) reactivity control, (2) reactor core cooling and heat removal from the primary system, (3) reactor coolant system integrity, (4) radioactivity control, and (5) containment conditions. The design of the SPDS shall be integrated with the design of instrument displays based on Regulatory Guide 1.97 guidance and the development of function oriented emergency operating procedures (EOPs).

A set of the five types of variables specified in Regulatory Guide 1.97 is available in both the TSC and the EOF. In addition, all sensor data and calculated variables not specified in Regulatory Guide 1.97 but included in the data sets for the SPDS will also be available for display in both emergency response facilities. This will help the TSC and EOF to make severe accident management decisions. However, under some accident conditions, such as that in a station blackout sequence, some plant instrumentation information that may help in severe accident management could be lost. Contingency plans for obtaining plant information from local instrument taps rather than remotely in the control room may be of benefit in this case.

The three qualification categories are defined in Position 1.4 of Regulatory Guide 1.97 as follows: "In general, Category 1 provides for full qualification, redundancy, and continuous real-time display and requires onsite (standby) power. Category 2 provides for qualification that is less stringent in that it does not (of itself) include seismic qualification, redundancy, or continuous display and requires only a high-reliability power source (not necessarily standby power). Category 3 is the least stringent. It provides for high-quality commercial-grade equipment that requires only offsite power." For both Category 1 and 2 variables, the instrumentation should be qualified in accordance with Regulatory Guide 1.89 [29]. There is no specific provision for the qualification of Category 3 equipment.

The environmental qualification of the Category 1 and 2 equipment includes consideration of temperature, pressure, humidity, and radiation conditions. It also accounts for the effects of sprays and chemicals. The environmental profiles described in IEEE Std 323-1974 [30] are acceptable to Regulatory Guide 1.89 [29]. They are based on the postulated design basis accident event (LOCA events) with additional margins to cover uncertainties. The margins required for the qualification curves are: an increase of 15°F for the temperature profile, an increase of 10% gauge pressure for the pressure profile, and an increase of 10% in the time period the equipment is required to be operational. IEEE Std 323-1974 calls for qualification for a typical integrated radiation dose of 26 Megarads and a spray exposure of demineralized water at a rate of 0.15 gal/min/ft^2. The instruments outside the primary containment are qualified for the expected environmental conditions, which may be less severe than those within the primary containment and are plant specific.

Instruments whose ranges extend beyond the qualification values specified in IEEE Std 323-1974 are required by Regulatory Guide 1.97 to follow the guidance provided in ANS-4.5 [31] for equipment qualification: The value of the maximum range, instead of the value obtained from the design basis accident events, of the monitored variable is to be used as the peak value in the qualification profile. Only the qualification profile of the measured variable needs to be extended and the other profiles remain as those derived from design basis accident events. The environmental qualification of the containment pressure instrument for detecting potential containment breach is an example: While the peak value obtained from design basis accident events is about the design pressure, the required instrument range is four times the design pressure (for a steel containment). This instrument is therefore qualified for a pressure of four times design pressure. However, the qualification temperature is still that from design basis accident events.

The availability of an instrument during a station blackout sequence depends on its power supply and seems to be plant specific. In general, all control room instrument information will be lost after the depletion of all station batteries. Since station blackout (SBO) contributes significantly to the total core damage frequency for Grand Gulf, lack of instrument indication during SBO presents a serious problem for CRM, particularly after the depletion of plant batteries. Methods to obtain plant status information without electric power need to be identified. For example, drywell temperature information could be available at indicators accessible from outside the control room, suppression chamber and drywell temperature information can be obtained by monitoring installed thermocouple elements using a portable self-powered potentiometer, and containment pressure information may be available from mechanical pressure gauges. The plant information that is not readily available in the control room but can be obtained elsewhere in the plant during station blackout will be plant specific. It is important to identify the availability of, means to access, and manpower required to collect information not readily available in the control room. An independent power supply for plant parameters that are important to CRM such as that required by CPI for RPV depressurization may also be desirable.

Table 3.1 Comparison of Mark III Containments (Reference 17)

Plant Name	Drywell Material & Construction	Drywell Free Volume (ft^3)	Drywell Design Temp. (deg F)	Containment Material & Construction	Containment Min. Free Vol. (ft^3)	Suppression Pool Min. Water Vol. (ft^3)	Containment Design Temp. (deg F)	Containment Design Pressure (psig)	Design Leak Rate % vol/day
Clinton	Reinf. Conc.	250,000	330	Reinf. Conc. W/Steel Liner	1,075,300	135,700	185	15	0.65
Grand Gulf	Reinf. Conc.	270,000	330	Reinf. Conc. W/Steel Liner	1,264,000	136,000	185	15	0.35
Perry	Reinf. Conc.	278,000	330	Steel	802,000	120,000	185	15	0.2
River Bend	Reinf. Conc.	251,000	330	Steel	713,000	127,930	185	15	0.26

Table 3.2 Mark III Containment Design Characteristics (Reference 13)

	Perry	River Bend	Grand Gulf	Clinton
Rated Thermal Power, MWt	3,579	2,894	3,833	2,894
Drywell Design Pressure, psig	+30.0 -21.0	+25.0 -20.0	+30.0 -21.0	+30.0 -17.0
Containment Design Pressure, psig	+15.0 -0.8	+15.0 -0.6	+15.0 -3.0	+15.0 -3.0

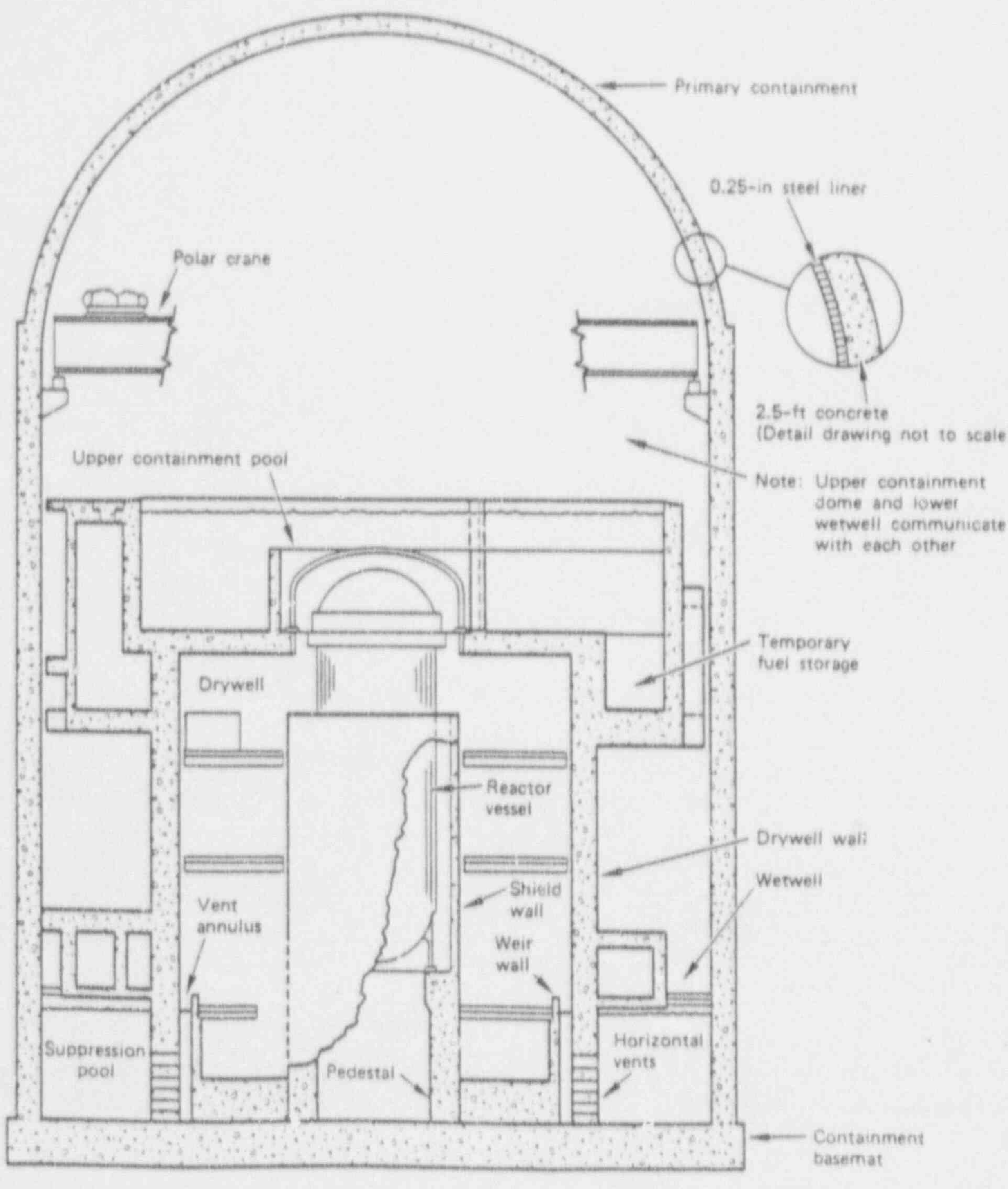


Figure 3.1 Grand Gulf Primary Containment (Referenced from NUREG/CR-4242)

4 Strategy Identification

The strategy identification process used in this report is the same as that discussed in a previous report on Mark I containments [32]. Existing information on severe accidents is reviewed to identify (1) the challenges a Mark III containment could face during the course of a severe accident, (2) the mechanisms behind these challenges, and (3) the strategies that can be used to mitigate these challenges. A systematic method utilizing a simplified event tree structure is employed to guide the review effort. One result of this examination is a safety objective tree which presents in a tree structure 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 to counter these mechanisms and thus mitigate the effects of the challenges.

In the following sections, the containment and release event tree (CRET) used for strategy identification is briefly discussed. (A more detailed discussion can be found in the Mark I report [32].) This is followed by a discussion of the challenges and strategies identified by the process and a presentation of the safety objective tree which summarizes the results of this identification effort (Figure 4.6).

4.1 Containment and Release Event Trees

The containment and release event trees (CRETs) used in the present investigation are simplified containment event trees covering the different phases of a severe accident. Each CRET covers a time period of distinct plant status characteristics and distinctive emphasis of severe accident management (SAM) activities. The early CRET extends from the beginning of an accident, up to the time when the reactor pressure vessel (RPV) breaches. Procedures based on existing EPGs are expected to be applicable and carried out during the early part of this period before significant core degradation occurs. In-vessel severe accident management activities to prevent core damage, or retain the core in the RPV if core damage is unavoidable, will be emphasized during this time. The late CRET covers the time period between vessel breach (VB) and containment failure (CF). The primary objective of SAM activities during the late CRET is to maintain containment integrity. The release CRET covers the time period after containment failure. Here the emphasis of SAM activities is to minimize the consequence of offsite fission product releases. Since containment failure could occur in any phase of an accident, procedures based on existing EPGs or in-vessel activities may be carried out concurrently with release management activities. Figure 4.1 shows the time phases of accident progression, as well as the time phases covered by the CRETs and the accident management guidelines (including the existing EPGs).

Besides being used for challenge and strategy identification, the CRETs could also be used to quantify the risk reduction offered by the strategies, and as a severe accident management tool for accident management decision making. These aspects of the CRET have been discussed in the Mark I report [32].

4.2 The Identification of Challenges, Mechanisms, and Strategies

The CRETs are used to examine some important accident sequences to identify the challenges, the mechanisms behind these challenges, and the strategies which can mitigate these challenges.
Figure 4.2 shows the containment fission product (FP) release profiles for Grand Gulf as presented in NUREG-1150 [4]. It shows the conditional probabilities of the accident progression bins (APBs) for the plant damage states (PDSs) that contribute significantly to the total plant core damage frequency (CDF). Table 4.1 presents a more detailed list of the most important PDSs for Grand Gulf, along with their mean core damage frequencies (MCDFs) and their percentages in the total mean core damage frequency (TMCDF). As shown in Table 4.1, fast station blackout (or short term station blackout, STSB, in Figure 4.2) contributes 94.3% to the TMCDF for Grand Gulf. Other important PDSs presented in Table 4.1 are the slow station blackout (or long term station blackout, LTSB, in Figure 4.2), the fast and slow ATWSs, and the fast and slow transients. In NUREG-1150, a fast accident scenario is defined as one with core damage occurring in a short time after accident initiation (approximately 1 hour), and a slow accident scenario is defined as one with core damage occurring in the long term after accident initiation (approximately 12 hours).

Table 4.2 shows the timing of key events for accident sequences associated with the above PDSs and other sequences such as the loss of containment heat removal sequence (TPI in Table 4.2). Table 4.2 shows that containment failure could occur at different times in different sequences. The values shown in the table are from calculations by the Source Term Code Package (STCP) [10, 11] and are typical for accident progression without any operator intervention.

The accident sequences discussed above have been examined in the challenge and strategy identification process. The challenges, mechanisms, and strategies identified in the various time phases of a severe accident are discussed in the following. The important time phases, as shown in Table 4.2 and Figure 4.1, include the very early phase, before significant core melt has developed; the early phase, between the end of the very early phase to slightly after vessel breach; the late phase, when the core debris is discharged to the reactor cavity and core-concrete interaction (CCI) is in progress; and the release phase, when containment integrity is lost.

For cases where similarity exists between a Mark III and a Mark I containment, only brief discussions will be presented in this report. More detailed discussions for these cases can be found in the Mark I report [32].

4.2.1 The Very Early Phase

The challenges to containment integrity during the very early phase, before significant core melt has developed, include suppression pool (SP) boundary loads and containment pressure loads. As discussed in the Mark I report, the mechanisms that may cause significant SP boundary loads include (1) SRV air clearing with higher than normal SP water level and (2) SRV steam condensation, and (3) vent chugging with low noncondensible gas content in the drywell. The mechanisms that cause significant containment pressure loads include (1) loss of pressure suppression capability either due to high SP temperature or SP bypass and (2) inadequate containment heat removal (CHR). The drywell temperature may exceed its design value in some accident sequences but it will not reach a value that challenges containment integrity in this very early phase.

Existing EPGs [15] are expected to be applicable during this phase of an accident. The control variables in the primary containment control guideline include SP temperature and water level, containment pressure, and drywell and containment temperatures. When the value of a control variable exceeds its predefined limit the operator is instructed to use designed plant features, e.g., the primary containment cooling systems, to maintain it within limits. If this effort is not successful, the operator will then take additional actions to mitigate the effects of this abnormal plant condition. The Grand Gulf EOPs also instruct the operator to turn on the hydrogen ignition system (HIS) when the RPV water level drops below the top of active fuel (TAF).

The mechanisms of the challenges and the strategies to mitigate these challenges are similar to those discussed in the Mark I Report [32]. However, there are some differences that are worth noting and are discussed below.

In a Mark III containment, the suppression pool makeup (SPMU) system can add a large amount of water to the suppression pool in a short time. In Grand Gulf the upper pool dump can increase the SP water level by five feet and significantly increase the length of water slug in the SRV line. The longer water slug could cause larger dynamic loads on SP boundary and structures, as well as larger loads on the SRV line, the SRV discharge quencher device, or its supporting structures. Of the other SP boundary loads mentioned above, the chugging load is a concern primarily in a LOCA event, which is not a significant severe accident sequence for a Mark III containment (Table 4.1).

Suppression pool bypass is not a significant concern for a Mark III containment. The nominal leakage area between the drywell and the wetwell for Grand Gulf is 0.017 ft^2 [7]. This is larger than the corresponding value in a Mark I containment but is within the technical specification limit and is not a SP bypass concern. The probability of having a pre-existing bypass area greater than the nominal value but still within technical specifications is estimated to be 0.04% and the probability of having a pre-existing large bypass area that would prevent vent clearing during slow drywell pressurization is estimated to be zero in NUREG-1150 analysis [7].

Furthermore, in all the important PDSs identified in NUREG-1150 (Table 4.1) the discharge of the RPV inventory is through the SRV lines to the suppression pool during this phase of an accident. This makes the SP bypass issue even less important. The actions specified in the BWR EPGs should be adequate for addressing any problem caused by a SP bypass.

Similar to that for a Mark I containment, overpressurization for a Mark III containment could occur during this time phase due to inadequate containment heat removal (CHR) capability. This may occur in ATWS sequences, where containment energy input exceeds the designed CHR capability, and in TPI sequences (TW sequences for Mark I), where the designed CHR capability is not available (Table 4.2). Containment overpressure may also occur in a Mark III containment during a very long-term SBO. For a case with RCIC injection, a NUREG-1150 calculation indicates that the containment pressure will be 498 kPa (72 psig) after 18 hours [7, 8]. This is greater than the mean containment failure pressure of 383 kPa (55 psig) used in the NUREG-1150 analysis [7]. Containment venting can be used to relieve containment pressure and prevent containment failure. Containment venting in a Mark III containment, unlike that in a Mark I containment, should not cause the failure of the ECCS pumps taking suction from the SP, because the ECCS pumps in a Mark III containment have been designed to pump saturated water [4, 8]. The containment overpressure challenge for a Mark III containment can also be mitigated by an upper pool dump. The large amount of water in the upper containment pool (36,380 ft^3) can be dumped in approximately 7.5 minutes through one of two dump lines [22] and provide additional heat capacity to the suppression pool for energy absorption.

Table 4.3 summarizes the challenges, mechanisms, and strategies during the very early phase of an accident.

4.2.2 The Early Phase

The early phase of a severe accident covers the time period between the onset of core melt to shortly after vessel breach (VB). This phase is characterized by increasing radioactivity and hydrogen gas in the containment atmosphere. The primary containment area radiation and hydrogen concentration monitoring systems can provide the information needed to deduce core damage. Additional information such as those from in-vessel instrumentation or other area radiation monitoring systems can also provide useful diagnostic information. This phase is further divided into two time periods, the time period before vessel breach and the one after vessel breach. The challenges, mechanisms, and strategies in the two time periods are discussed below.

4.2.2.1 Before Vessel Breach

The challenges to containment integrity during this time period include the SP boundary load due to SRV actuation and the containment pressure and temperature loads due to hydrogen generation and combustion.

Suppression pool boundary load due to SRV actuation will occur if the RPV remains at high pressure during core degradation. Since the mass of noncondensible gases discharged into the SP is much greater than that originally in the SRV discharge line, used as the basis for the design assessment load [33], the SP boundary load from SRV actuation after core melt will be different, and may be greater, than the design assessment load. This SRV air clearing load will add to the containment pressure, and the combined load may threaten containment integrity. Since the SRV air clearing load is caused by high pressure SRV actuation, the load can be mitigated by keeping the RPV pressure low.

The containment pressure increase due to mass and energy addition to the containment atmosphere during this time period is not expected to cause significant increase in containment failure probability. The mass and energy addition arises from the heat and gases generated in the RPV from decay heat and fuel cladding oxidation (principally zirconium oxidation in this time phase). Because of the large volume of a Mark III containment, the amount* of hydrogen released to the containment during core degradation is smaller than the amount of

*As used in this report refers to Kg-moles of gas.

noncondensible gases originally in the containment, and thus will not result in a significant pressure increase in the containment. For Grand Gulf, the hydrogen generated from 50% zirconium oxidation is about 870 kg-moles. This is less than 50% the amount of air originally in the containment (1,800 kg-moles) and will cause a pressure increase of about 7 psi if containment temperature is about normal. The temperatures in the containment and drywell atmosphere are expected to be generally low, below their respective design values (330 °F for the drywell and 185°F for the containment), for important severe accident sequences in a Mark III containment [7, 10, 11] and should not contribute significantly to containment pressure rise and failure probability.

Hydrogen combustion is the most important loading condition during this time period. Hydrogen concentration in the containment atmosphere is influenced by the amount of zirconium oxidized during core degradation. With 10% zirconium oxidation the amount of hydrogen generated in the RPV will be about 170 kg-moles for Grand Gulf, or about 10% the amount of noncondensible gases originally in the containment. The zirconium oxidation level is in general predicted to be above 30% by the STCP [10, 11] or MELCOR computer codes [9] and less by the MAAP computer code [14]. The mean values for the total amount of hydrogen released during core degradation used in the NUREG-1150 analyses, provided by the In-Vessel Phenomenology Expert Panel, ranges from 222 to 466 kg-moles [7], or 12% to 26% the noncondensible gases originally in the containment. Containment hydrogen concentration could therefore reach detonation level (hydrogen concentration greater than 16%) during this time period.

Figure 4.3 presents the mean probabilities of containment and drywell failure before vessel breach predicted in NUREG-1150 [7]. It shows that both the drywell and the containment could be failed due to a hydrogen burn (in either the deflagration or the detonation mode) before vessel breach. Figure 4.3 also shows that a hydrogen burn is the most important cause for containment and drywell failure before vessel breach. It should be noted, however, that these failures are caused by hydrogen burns in the containment, not in the drywell. Hydrogen concentration in the drywell before VB is expected to be small because the RPV inventory is discharged to the containment through the suppression pool for important severe accident sequences (Tables 4.1 and 4.2). Hydrogen can enter the drywell only through a stuck-open SRV tailpipe vacuum breaker or via in-leakage through the drywell wall. A stuck-open SRV tail pipe vacuum breaker may cause a flammable condition to exist in the drywell atmosphere for a short time (e.g., 20 minutes for a SBO sequence [9]) before the drywell is inerted by either steam buildup or oxygen depletion. A hydrogen burn in the drywell, even if it does happen, is not likely to challenge containment integrity [9].

The combustible gas control system provided in Grand Gulf includes (1) the drywell purge system, which is designed to purge the hydrogen produced within the drywell into the larger containment volume, (2) the containment purge system, which is designed to purge the containment atmosphere through filter trains and to take outside air as make up through a compressor into the containment, (3) the hydrogen recombiners, which are designed to control long-term containment hydrogen concentration in a LOCA event, and (4) the hydrogen ignition system (HIS), which is designed to provide distributed ignition sources throughout the containment and the drywell to burn the hydrogen in a controlled manner. The first three systems are designed primarily for post-LOCA hydrogen production and may not have sufficient capacity to handle the amount of hydrogen produced in a severe accident.

At Grand Gulf, the compressor of the drywell purge system forces the containment atmosphere to the drywell. Continued operation of the compressor will then cause the drywell atmosphere to flow through the suppression pool to the containment. This causes the hydrogen in the drywell produced in a LOCA event to be transported to the much larger volume of the containment. However, when the hydrogen produced in-vessel in a severe accident is discharged to the containment, the operation of the drywell purge system will increase the hydrogen concentration in the drywell and thus increase the probability of a hydrogen burn in the drywell. The use of this system in a severe accident may not be desirable and may need to be avoided.

The containment purge system can be used to reduce containment hydrogen concentration following a design basis LOCA. The hydrogen rich containment atmosphere is discharged to the outside and replaced by the outside air. However, the containment purge fans have a very low pressure head (4 inches of water) and limited capacity

(3,000 cfm for each of two fans), and are not effective in supplying outside air to the containment to change its composition during a severe accident when containment pressure is high and the hydrogen production rate is rapid. As a result, the containment purge system is not effective for reducing the threat of hydrogen deflagration/detonation in short-term station blackout sequences [13]. Nonetheless, the purge system can be used to vent the containment to reduce its pressure and the amount of gases it contains. Containment venting can thus reduce the impact of a hydrogen burn by reducing the containment base pressure and the amount of hydrogen burned. On the down side, hydrogen burns during containment venting will result in a significant driving force for the release of the containment atmosphere and the fission products it contains.

Containment venting during a long-term sequence when the containment is steam inerted can remove oxygen, along with other containment gases, from the containment. A hydrogen burn later in the accident, after steam is condensed and the containment de-inerted, may be avoided if oxygen concentration is sufficiently low. However, if sufficient amounts of noncondensible gases are removed from the containment, later steam condensation may cause the pressure in the containment to be less than the atmospheric pressure. This will result in a negative pressure loading on the containment as well as an influx of oxygen to the containment, causing a flammable mixture to form in the containment again. Negative containment pressure and oxygen influx may not occur if there is still a sufficient amount of noncondensible gases (e.g., hydrogen) remaining in the containment after containment venting. The use of containment venting for this purpose therefore requires careful monitoring and control.

The hydrogen recombiner system is designed to control the long-term hydrogen buildup in the containment from radiolysis in a design basis LOCA event, which proceeds at a much slower rate than the metal-water reaction in a severe accident. The recombination rate of 100 cfm is too low for the expected hydrogen production rate in a core degradation event, and the recombiner may become an unwanted ignition source when the hydrogen concentration reaches the flammable limit.

The hydrogen ignition system (HIS) is the most effective system to control the hydrogen produced in a severe accident. It is designed to burn the hydrogen in such a manner that containment failure from a hydrogen burn will not occur. However, the existing HIS relies on ac power and will not work during a station blackout (SBO) sequence. Backup power to the ignitors, or the use of catalytic ignition systems, has been suggested to improve the performance of the HIS in SBO sequences [12]. Since short term SBO sequences contribute to over 90% of the total PDS for Grand Gulf, these improvements would have a significant impact on risk reduction. These improvements are less effective for long term sequences because of the potential of steam-inerting in these sequences.

There are potential adverse effects of using the HIS. For example, the operation of the HIS may start a diffusion flame above the SP at the location of the discharging SRV quencher device. A continuous diffusion flame may cause an overtemperature failure of adjacent equipment such as the elastomeric seals in both the containment and the drywell and the wetwell-to-drywell vacuum breakers. However, this has been shown to be an unlikely failure mode for Grand Gulf [12, 34].

As discussed in the Mark I report, the containment spray is a very important system for severe accident management. The containment spray may mitigate hydrogen combustion by the following mechanisms: (1) Spray droplets on the order of 20 microns or less in diameter can significantly raise the lower flammability limit for hydrogen combustion; (2) sprays can enhance cooling of the burned gases and therefore cause pressures and temperatures to decrease more rapidly to precombustion levels; and (3) water sprays may have the potential of reducing the probability of detonation [35]. However, there is considerable uncertainty in droplet size and fog density resulting from fog formation in the containment during severe accidents. Moreover, large-water-droplet sprays tend to increase flame speeds by promoting mixing in lean hydrogen-air mixtures and cause peak pressures to be closer to the adiabatic, constant-volume values. Furthermore, containment sprays will change the containment atmosphere composition by removing steam (a diluent) from the containment atmosphere and may thus increase the probability of hydrogen burns. Since the containment is inerted by steam in some of the long-term severe accident sequences for a Mark III containment, the use of containment sprays can result in containment de-inerting and should be carefully controlled. Since the primary function of containment spray is to reduce the pressure rise, it can be used with the HIS to reduce the pressure rise from the benign hydrogen burns initiated by the HIS.

Since there is no ignition source in the Grand Gulf containment during a station blackout, hydrogen burn may not occur even when the containment atmosphere is flammable, because of the lack of an ignition source. Actuation of the HIS, either manually or after a power recovery, will provide the required source for hydrogen burn, and containment failure may occur if the containment atmosphere has reached the deflagration/detonation limit. It is therefore important that the HIS be turned off during a station blackout, and that the power to the HIS be restored after a power recovery only if it can be assured that a detrimental hydrogen burn will not occur. Existing plant procedures have provided such instructions to guard against initiating damaging hydrogen burns by the HIS.

The challenge of hydrogen combustion during this time period can also be mitigated by reducing the amount of hydrogen production before VB. This can be achieved by initiating RPV depressurization at the optimum RPV water level [12]. Analyses of short-term station blackout sequences show that the time to vessel failure is extended, and the amount of in-vessel hydrogen generation is reduced, if RPV depressurization is delayed from the time the RPV water level is at 71% core height (EPG Revision 4 requirement) to the time the RPV water level is at 33% core height (EPG Revision 3 requirement) [13].

Table 4.4 summarizes the challenges, mechanisms, and strategies for this time period.

4.2.2.2 After Vessel Breach

Figure 4.4 shows the drywell and containment failure probabilities at VB for Grand Gulf. Since the energetic events associated with VB occur in the drywell, they would challenge the drywell integrity first. Containment is challenged when the mass and energy generated in the drywell are transported to the containment through either a drywell break or the SP vents. Furthermore, both the drywell and the containment may also fail in the Alpha mode, and the discharge of the RPV inventory during a high pressure VB will also result in SP hydrodynamic loads similar to those occurring in a design basis LOCA event.

The loads that challenge the integrity of the drywell at vessel breach include the quasi-static pressure loads (rapid pressurization) associated with high pressure melt ejection (HPME) during vessel blowdown, the impulse pressure loads resulting from hydrogen detonation at VB, and the loads caused by steam explosions as the molten core debris interacts with the water in the reactor cavity. The quasi-static pressure load associated with the HPME and the impulse load from an ex-vessel steam explosion also challenge the structural integrity of the RPV pedestal and, indirectly, the integrity of the drywell.

After the initial rapid pressurization of the drywell, the gases in the drywell will be released to the containment and cause a pressure rise in the containment. Because of the large volume of the containment, the pressure rise should in general be moderate. However, the probability of hydrogen combustion in the containment will be increased after VB due to both an increase in containment hydrogen concentration and the availability of ignition sources provided by the ejected core debris. While a hydrogen deflagration results in a quasi-static pressure load on the containment, a hydrogen detonation generates an impulse load on the containment. Both will threaten the integrity of both the drywell and the containment.

The challenges at or immediately after VB could fail the containment during or shortly after vessel breach and result in significant fission product release to the environment. Since there may not be sufficient time for plant operation personnel to take mitigating actions, any strategies must be carried out before vessel breach to be effective. The mechanisms and strategies related to these challenges are discussed below.

<u>SP Hydrodynamic Loads</u>: Suppression pool hydrodynamic loads similar in nature to those occurring during the blowdown phase of a design basis LOCA event, e.g., pool swell [33], will occur after high pressure VB. The mass and energy additions associated with the blowdown of the primary system after VB are different from those of the

design basis LOCA event. Both the amount of noncondensible gas (i.e., hydrogen) and the temperature in the primary system can be higher than in a LOCA event. The loading conditions associated with suppression pool hydrodynamics due to the blowdown of RPV gases and molten core debris may challenge containment integrity when combined with other containment loads. Since the SP hydrodynamic loads are caused by a high pressure RPV blowdown, their effects can be mitigated by maintaining the RPV at low pressure before vessel breach.

Drywell Quasi-Static Pressure Loads (Rapid Drywell Pressurization): At VB, the gases and the core debris in the RPV are discharged to the drywell. This causes a rapid pressure rise in the drywell and creates a pressure differential between the drywell and the containment before a communication path between the two compartments (e.g., SP vents or drywell break) is established. The mechanisms that can cause rapid drywell pressurization and possible containment failure at vessel breach are (1) direct containment heating (DCH), (2) hydrogen burn at VB, (3) ex-vessel steam explosions (EVSE), and to a lesser extent, (4) mass and energy addition to the containment atmosphere at vessel breach.

The pressure increase due to DCH and mass and energy addition is significant only if the RPV fails at high pressure. Their effects can therefore be removed or reduced by RPV depressurization. DCH involves a series of physio-chemical processes that contribute significantly to the energy and mass input to the drywell. In DCH, a fraction of the ejected core debris may be dispersed into the containment as fine particles, and a substantial portion of the debris' sensible heat can be transferred rapidly to the atmosphere. The metal in the dispersed debris can react chemically with the oxygen or steam in the containment atmosphere (an exothermic reaction) and release more energy and noncondensible gases. The impact of DCH on containment integrity has many uncertainties. For example, the severity of DCH depends on the fraction of molten core ejected, the unoxidized metal content in the melt, the mode of vessel failure, and the timing for vent clearing. Since a large amount of aerosols, including refractory fission products, could be generated in HPME, a significant release of radioactive material could result should both the containment and the drywell fail due to the DCH loading.

In general, water in the reactor cavity before VB is believed to have a beneficial effect on DCH. However, the mitigating effect of water on DCH is still not clear. The water in the reactor cavity could either be dispersed ahead of the bulk of the ejected debris or co-dispersed with the debris. The water co-dispersed with the debris may continue to quench the debris and thus mitigate the effects of DCH. On the other hand, the steam generated in this process would increase containment pressure or cause additional metal oxidation. The effects of water on DCH are sensitive to the timing and location of water addition, the assumptions regarding droplet-debris reaction kinetics, and the amount of water involved. Additional research may still be needed in this area.

Hydrogen burn at VB will occur if the drywell atmosphere is combustible and a drywell hydrogen burn does not occur before VB due to lack of ignition sources. The hot gases and hot debris particles released from the vessel provide additional hydrogen and required ignition source for combustion. However, the drywell is most likely to be steam inerted at this time (as discussed above) and a hydrogen burn is not likely to occur even with the availability of ignition sources.

An ex-vessel steam explosion may occur when the molten core debris contacts water in the reactor cavity. The energetic and violent fuel coolant interactions (FCI) will cause a rapid pressure increase and an impulse load in the reactor pedestal and the drywell. Ex-vessel steam explosions cannot occur if there is no water in the reactor cavity. Water is introduced to the reactor cavity either from a failed vessel by core injection or from the SP by a high wetwell-to-drywell pressure differential. The pressure differential will depress the suppression pool water level and cause an overflow of the SP water to the drywell, from where the water is directed to the reactor cavity either through the drywell drains or through the reactor pedestal door. The amount of water to the drywell before VB depends on the containment pressurization rate (e.g., from hydrogen discharged to the containment through the SRVs and SP evaporation) and the leak area between the containment and the drywell. Degassing from the drywell concrete structure and vessel heat loss will increase drywell pressure and thus reduce the potential for weir wall overflow, and hydrogen combustion in the containment will increase containment pressure and thus increase the amount of water transported to the drywell. There are other actions that can be taken to reduce water overflow. The operation of the drywell-to-wetwell vacuum breakers will reduce the wetwell-drywell pressure differential and thus reduce the amount of water overflow, and an upper pool dump will increase SP water level and thus increase the potential for overflow. Containment venting can maintain the containment pressure low and is the most effective method to avoid a positive wetwell to drywell pressure differential.

Because the consequence of a large ex-vessel steam explosion may be very severe, the potential for reactor cavity flooding and means to reduce this potential have been discussed and analyzed. Figure 4.5 shows the impact of various mitigating actions on the drywell cavity water inventory at the time of VB for a Mark III containment during a short-term station blackout [13]. It shows that containment venting is the single most effective action in reducing the amount of water refluxed into the drywell. Reference 1? also predicts that no water is refluxed into the drywell if there is no hydrogen burn in the containment. It should be noted that the results are very sensitive to modeling assumptions. MELCOR analyses for Grand Gulf station blackout scenarios show that, even without a hydrogen burn, varying some important parameters within their uncertainty ranges would result in SP backflow in some cases, but no backflow in some other equally valid cases [9].

Figure 4.5 does not include the effect of an upper pool dump. An upper pool dump will add a large amount of water to the suppression pool in a short time. The SP water level could almost reach the top of the weir wall if the pool level before upper pool dump is near normal. The upper pool dump will therefore greatly increase the potential for cavity flooding. Avoiding upper pool dump before VB has been suggested as a mitigating strategy to reduce the probability of ex-vessel steam explosion [12]. One adverse effect of avoiding an upper pool dump is an increase in the probability of SP bypass.

RPV Pedestal Quasi-Static Pressure Load: The above mechanisms that cause a quasi-static pressure load on the drywell will also cause a quasi-static pressure load on the volume inside the RPV pedestal (reactor cavity). The volume of the reactor cavity is small (about 8,600 ft^3) and the communication path with the drywell volume is restricted (primarily by a door of 3 feet by 7 feet). The quasi-static pressure load in the cavity may be much larger than the corresponding load in the drywell. The strength of the pedestal is also greater than that of the drywell (an estimated failure pressure of 189 psig for the pedestal versus of 85 psig for the drywell) [7]. Failure of the pedestal would result in a gross motion of the RPV and fail the drywell boundary by damaging the drywell wall or the seals at piping penetrations through the drywell wall. The conditional probability of drywell failure used in NUREG-1150 analyses, given pedestal failure, is 0.175. Since the mechanisms for this load are the same as those for the drywell load, similar mitigation strategies apply.

Drywell Impulse Load: Hydrogen detonation will occur if the drywell mixture reaches the detonation limit. A hydrogen detonation will generate an impulse load on the drywell structure and may fail the drywell by a dynamic load. The probability of drywell failure by this load is small because the drywell is most likely to be inerted at this time.

Pedestal Impulse Load: In addition to contributing to the overall quasi-static pressure loads to the drywell and the reactor pedestal, an ex-vessel steam explosion will also generate an impulse load on the pedestal. There are significant uncertainties on the magnitude of this dynamic load and its effect upon the pedestal structure. To reflect these uncertainties, a uniform distribution between zero and one is used to determine the conditional probability of pedestal failure, given the occurrence of EVSE, in the NUREG-1150 analyses. The strategies discussed above for mitigating EVSE can be used to mitigate this challenge.

Containment Pressure Load: The drywell pressurization at VB discussed above is relieved either through the suppression pool vents or through a drywell break to the containment. The corresponding containment pressure increase is not expected to be large enough to challenge containment integrity even if the drywell has been ruptured. However, it does increase the containment base pressure and makes the containment more vulnerable to challenges that occur later, e.g., hydrogen burn.

The most significant challenge to containment integrity during this time phase is a hydrogen burn in either the deflagration or the detonation mode. The energetic events that occur in the drywell may provide additional hydrogen as well as ignition sources to the containment. A hydrogen burn may also be started by random ignition sources. In the NUREG-1150 analyses, hydrogen burn is assumed to be certain if ac power is available because there are numerous ac sources that are potential ignition sources. In cases where ac power is not available the probability of ignition will depend on the hydrogen concentration in the containment. The mean value of ignition probability varies from 0.49 for a hydrogen concentration greater than 16% to 0.21 for a hydrogen concentration between 4% and 8%. In both the TB2 and the TBS sequences in Table 4.2 the containment is failed by hydrogen burns at VB when an ignition source is assumed to be available.

Hydrogen deflagration will result in a quasi-static pressure load. This pressure load may fail either the containment or the drywell, or both. Because the pressure rise associated with this event is very rapid, the pressure differential load on the drywell is reduced, but not eliminated, even if the containment has already failed. In the NUREG-1150 analyses, a hydrogen deflagration is assumed to occur if the containment hydrogen concentration is above 6%, steam concentration below 55%, and oxygen concentration greater than 5%. A hydrogen detonation may occur if the hydrogen concentration is greater than 16% and the steam concentration less than 35%. A hydrogen detonation will result in an impulse load on both the containment and the drywell and may fail either or both of them.

The strategies to mitigate hydrogen burns after VB are the same as those before VB, discussed in Section 4.2.2.1. The most important system for controlling a hydrogen burn is the HIS. The containment purge system valves can be used to vent the containment to reduce its base pressure and the amount of the combustible gas mixture in the containment. However, without an effective compressor system to supply outside air to the containment the composition of the containment atmosphere cannot be changed. Containment venting can also remove a sufficient amount of oxygen from the containment such that a hydrogen burn will not occur even if the containment atmosphere is de-inerted later. It is also important to avoid ignition sources in the containment if the containment gas mixture is combustible, and to avoid deinerting the containment if the containment is steam-inerted. The containment spray may have some beneficial effects on hydrogen burns, but there are significant uncertainties, and de-inerting of the containment atmosphere by steam condensation may result in hydrogen burns and severe consequences.

<u>Alpha Mode Failure</u>: An energetic in-vessel steam explosion could fail the vessel and generate a missile that fails both the drywell and the containment. An alpha mode failure is not very likely in a BWR containment, and less likely if the RPV is at high pressure than at low pressure. A mean failure probability of 0.001 is used in the NUREG-1150 analyses for high RPV pressure, and 0.01 for low RPV pressure (less than 200 psi). The strategy to reduce a potential Alpha mode failure is therefore to maintain the RPV at high pressure. This contradicts other strategies that require RPV depressurization. Because of the low probability of an Alpha mode failure in a BWR containment and the benefit of RPV depressurization this strategy is not likely to be implemented.

4.2.3 The Late Phase

As defined in Figure 4.1, a severe accident enters the late phase after vessel breach but before containment failure. The sudden change in RPV and containment conditions associated with vessel breach indicates the beginning of the late phase. Failure of the RPV may result in a sudden increase in containment pressure and a sudden decrease in RPV pressure. The radioactivity in the drywell atmosphere may also show a sudden increase, since prior to vessel breach the discharge of the fission products to the containment atmosphere was through the SRV lines and the suppression pool, while after vessel breach the discharge goes directly to the drywell atmosphere. This increase in radioactivity will depend on the specific scenario and is likely to be more pronounced for transients than for LOCA's.

For the accident sequences presented in Table 4.2, containment failure is predicted to happen in the late phase for TB1 and TQUV sequences. Severe hydrogen burns do not occur in the calculation of these sequences. Hydrogen burns would occur and fail the containment if an ignition source is available at VB (i.e., TB2 in Table 4.2). Detrimental hydrogen burns do not occur in TQUV because ac power is available and the HIS is operating. The containment pressurization in these sequences is due to the buildup of steam and noncondensible gases in the containment atmosphere. Noncondensible gases are generated from both in-vessel core degradation and ex-vessel core-concrete interaction (CCI). This slow pressurization process may fail the containment, but is unlikely to fail the drywell because there is sufficient time to establish a pressure equilibrium across the drywell wall. A pressure differential across the drywell wall may exist at containment failure when rapid containment depressurization occurs. However, this pressure differential cannot be greater than the containment failure pressure, which is much smaller than the drywell failure pressure, and is not likely to challenge drywell integrity.

Containment cooling can be used to reduce the containment pressure if containment temperature is high, or if there is a large amount of steam in the containment. The containment spray is very effective in achieving this purpose, particularly if the RHR heat exchangers are functioning. However, containment spray may de-inert the containment if the containment is steam inerted.

Containment cooling cannot prevent a continuous containment pressure rise if the release of noncondensible gases continues. Containment venting is needed to remove the noncondensible gases and maintain the containment pressure below the failure pressure. Since the containment atmosphere is highly contaminated at this time, containment venting will cause the release of fission products to the environment. The suppression pool will provide fission product scrubbing, except for noble gases, if the drywell remains intact. Noble gases will be released to the environment during containment venting.

CCI is the most important mechanism for slow containment pressurization in the late phase. As the high temperature core debris falls into the reactor cavity after vessel breach, the molten core debris starts to heat and decompose the structural concrete. The steam, carbon dioxide and other oxidants released from the decomposing concrete will react with the metallic constituents in the corium and generate a significant amount of noncondensible and combustible gases and release the chemical heat of reaction into the corium pool. The release of the high temperature gases to the drywell atmosphere not only causes a pressure load on the containment but also a temperature load on drywell structures. The drywell temperature load is further augmented by the transfer of heat from the hot corium to the drywell atmosphere. The drywell atmosphere temperature has been predicted to be over 1,000°F [11, 13] during CCI. Drywell cooling, if available, can be used to mitigate the drywell temperature load. The drywell purge system in Grand Gulf can be used to transport the cooler containment atmosphere to the drywell and thus reduce the temperature load.

The progress of CCI is influenced by many uncertainties, e.g., the composition and mass of the core debris discharged from the RPV, the initial temperature and the decay heat level of the debris, the composition and material properties of the structural concrete, and the availability of water in the reactor cavity. Concrete erosion in the reactor cavity may weaken the reactor pedestal and result in vessel motion including tearout of piping penetrations through the drywell wall and consequently drywell failure. The downward progression of the concrete erosion may result in basemat melt-through and subsequent fission product release. Both of the above failure modes are expected to take a considerable time to occur and analyses show that the likelihood of this mechanism of failure is small for the BWRs analyzed, in part because other mechanisms are likely to result in failure earlier in the accident. For Grand Gulf, the geometrical arrangement of the reactor cavity is such that the weakening of the reactor pedestal is not severe even with long term and extensive concrete erosion [13]. The depth of the basemat of the containment, directly under the vessel, is about 11 feet and is unlikely to be penetrated before the occurrence of other failure modes.

Availability of water in the reactor cavity before vessel breach and a continuous supply of water to the reactor cavity after vessel breach are the most effective means to control the progress of CCI. CCI will not occur if the core debris is in a coolable configuration and there is water in the reactor cavity to cool it. The availability of water to the core debris at vessel breach will increase the probability of having the core debris at a coolable configuration. Water can be introduced to the reactor cavity before vessel breach by containment pressurization. Actuation of an upper pool dump before vessel breach will increase the probability of suppression pool water overflow to the drywell and consequently the availability of water in the reactor cavity. However, as discussed above, water in the reactor cavity increases the probability of ex-vessel steam explosions, and an upper pool dump, which results in water falling through a portion of the containment atmosphere, will condense the steam in the containment and increase the probability of hydrogen burns.

A continuous supply of water to the reactor cavity can keep the core debris flooded and cooled. The source of a replenishable water supply to the reactor cavity is any system that can inject water to the RPV. The water that is added to the RPV after vessel breach will flow through the break to the reactor cavity.

The reactor cavity can also be flooded by continuously adding water to the suppression pool to overflow the weir wall. The SP water level is almost at the top of the weir wall after an upper pool dump if the suppression pool water level is at its normal level before upper pool dump. Continuously adding water to the SP using an alternate water supply, such as that from the service water cross-tie, will result in SP water overflow and reactor cavity flooding.

Containment flooding can be accomplished by continuously adding water to the containment using external water sources other than the SP. Containment flooding is part of the RPV control guideline in the BWR EPGs (Contingency #6) [15], and as such it may be initiated early in an accident (as compared with most of the containment strategies discussed here). To flood the containment, the EPGs call for the suppression pool makeup system to rapidly add a large quantity of water to the containment, and for all available systems that take suction from sources outside the containment to deliver water to the containment. Contingency #6 of the EPGs describes in detail the systems, water sources, and procedures to be used to fill the containment.

The purpose of containment flooding, as required by Contingency #6, is to provide cooling to the reactor core and thus prevents RPV failure. Containment flooding, if not executed before VB, can be carried out later in the accident to provide corium cooling and FP scrubbing. Once accomplished, this strategy will serve to stabilize the accident with a minimal use of active systems.

The time needed for containment flooding depends on the level of flooding required and the water supply systems available at the time of the accident. It requires about 50,000 ft^3 of water to flood the containment up to the top of weir wall (about 25 ft from the bottom of the SP) for Grand Gulf. Consequently, it takes about 4 hours for a fire water system (typical capacity 1,500 gpm), or about 40 minutes for an emergency service water system (typical capacity 10,000 gpm) to supply this amount of water to the containment.

More water is needed to flood the containment up to the bottom of the RPV. For Mark III containments, each additional foot of water above the top of the weir wall will require about 10,000 ft^3 of water supply. The Grand Gulf EOPs allow the containment to be flooded up to a water level of about 85 ft. If the level reaches 85 ft, all external water supplies should be terminated.

The 85 (actually 84.75) ft. maximum containment water level is approximately at the RPV normal water level and almost reaches the flat roof of the drywell. The gases originally in the drywell (before containment flooding) will be trapped in the drywell and will be compressed as the drywell water level increases. Since the drywell atmosphere is compressed at a greater rate than the containment atmosphere, for each unit of water level increase the water level in the drywell will be lower than that in the containment as the containment is flooded. The water level difference depends on the initial containment pressure and the leak rate from the drywell to the containment. However, the drywell water level will reach the top of active fuel (TAF) level (approximately 20 ft below the normal water level) as the containment reaches its maximum level.

During containment flooding, the free volume of the containment is reduced and the containment atmosphere is correspondingly compressed. Because of the large size of the containment (1,670,000 ft^3 for Grand Gulf), the reduction in containment free volume for a containment flooding up the top of the weir wall will not be significant. However, significant free volume reduction occurs when the containment water level reaches its maximum allowable level of 85 ft. The containment pressure will correspondingly be increased and containment venting can be used to maintain or reduce containment pressure during containment flooding.

Mass and energy will be continuously added to the containment air space from decay heat and CCI, and containment pressure will continue to rise. It is important that the containment vent paths are not flooded as they are needed for containment pressure control. Since the core materials are submerged, the release through the drywell vents will have been subjected to pool scrubbing.

The release of combustible gases from CCI increases the probability of hydrogen burns in the containment. The strategies discussed above for combustible gas control during the early phases of a severe accident can also be used during this phase. Adding water to the reactor cavity can mitigate the progress of CCI and thus the production of combustible gases in the containment. Table 4.5 summarizes the challenges, the mechanisms, and the strategies for this phase of a severe accident.

4.2.4 The Release Phase

The accident enters the release phase when the containment loses its integrity and the containment atmosphere is discharged outside of the primary containment. This phase is characterized by high radiation outside the containment. The radioactivity release control guidelines of the BWR EPGs would have been initiated to control fission product release. The general aim in the EPGs of isolating the leak area, or isolating the leaking systems, is certainly applicable for release control during this accident phase, but the actual situation in a severe accident will most likely be much worse than that anticipated in the EPGs. Additional strategies beyond the existing EPGs are therefore beneficial to mitigate fission product (FP) release after containment failure.

Important factors that affect fission product (FP) release after containment failure include the amount of fission products in the containment atmosphere and the driving force for fission product release, i.e., containment pressure. The source of fission product release before vessel breach (in-vessel release) is the degraded reactor fuel. In general, almost all of the noble gases are released, and significant fractions of the more volatile radionuclides (I and Cs groups) will also be released. The release of Te is more uncertain but is expected to be significant, while the release of other less volatile FP groups will be small.

The fission products from in-vessel release will pass through the suppression pool (by SRV actuations) before entering the containment atmosphere as long as RCS integrity is maintained, and as a result, a significant fraction of the non-noble gas fission products will be retained in the suppression pool (suppression pool decontamination). Direct release to the drywell is possible if the vacuum breaker on the SRV tailpipe fails open. A stuck-open vacuum breaker will cause the drywell pressure to increase and a transfer of the drywell atmosphere to the containment through the suppression pool vents. Since the decontamination factor for flows passing through the suppression pool vents is smaller than that for flows passing through the SRV spargers, more fission products will be transported to the containment if there is a stuck-open vacuum breaker. Some of the fission products in the drywell will pass to the containment directly, bypassing the suppression pool, through the (normal) leak areas between the drywell and the wetwell. In cases where there has been a drywell failure, the suppression pool may be bypassed completely.

It is desirable to have the in-vessel release passing through the suppression pool, preferably through the SRV spargers, before discharging to the containment. RPV depressurization before VB will assure a discharge of RPV inventories through the SRV spargers and thus the greatest degree of decontamination achievable. The release of fission products to the containment can also be reduced by reducing the probability of stuck-open SRV tailpipe vacuum breakers. This can be achieved by using different SRVs for RPV pressure control (thus avoiding cycling of the low set SRV) and by extending the period of valve opening for each actuation (thus reducing the number of actuations required).

At vessel breach, the fission products in the RPV will be released first to the drywell and then to the containment. Suppression pool decontamination (through horizontal vents) will be maintained if drywell integrity is maintained. Significant aerosol generation and fuel fragmentation may occur if the vessel is breached at high pressure. The rapid pressure rise in the drywell during HPME will cause a rapid clearing of the SP vents and a direct release of the drywell atmosphere to the containment, without the full benefit of normal suppression pool decontamination. The fission products transported to the containment may therefore be increased significantly, and RPV depressurization before vessel breach is desirable. An ex-vessel steam explosion, if it occurs, may also cause a significant FP release to the containment, for reasons similar to those discussed above for HPME. EVSE can be avoided by eliminating water in the reactor cavity as discussed above.

The fission product release after vessel breach is primarily from CCI (ex-vessel release). The strategies that were discussed above for mitigating the progress of CCI can be used here to reduce ex-vessel fission product release. The flooding of the reactor cavity or the containment will not only mitigate the progress of CCI, but also provide pool scrubbing for the fission products released from CCI.

Besides CCI, fission products may also be introduced to the drywell after vessel failure by the heat up and revolatilization of the fission products deposited on the surfaces during core degradation. FP revolatilization is affected by post-vessel-failure thermal hydraulics, RCS heat transfer, and the chemistry of the retained radionuclides. Extensive RCS retention during the in-vessel release, high temperature of the RCS structures, and high flow rates inside the RCS after vessel failure all contribute to greater FP revolatilization. High drywell temperature will also promote FP revolatilization by reducing RCS heat removal. FP revolatilization will be reduced if the temperatures in the RCS and the drywell are kept low. Adding water to the vessel and initiating drywell cooling may achieve this objective. Adding water to the RPV will also scrub fission products from the RCS and thus reduce their release. Containment flooding up to a level that keeps a large part of the RPV submerged will reduce FP revolatilization from the RCS by maintaining a low temperature and providing pool scrubbing.

Another source of fission product release is the late release of iodine from the suppression pool or the water pools in the reactor cavity or the drywell floor. It represents a long term challenge to release control. Release of iodine from a water pool could be caused by (1) pool flashing at containment failure, (2) pool boiling as a result of decay heating, and (3) a change of the chemical form of the iodine in the pool. Other important factors affecting iodine release include the pool pH value and the radiation dose rate. In general, elemental iodine could be converted into nonvolatile forms of iodine by radiation in a pool at higher pH values.

Late release of iodine from water pools is influenced by the temperature and the pH value of the pool water. SP cooling, if available, can be used to keep the pool temperature below the boiling point and thus reduce the release of iodine from the SP. Adding water to the suppression pool, e.g., by upper pool dump or containment spray using alternate water sources, can dilute iodine concentration in the suppression pool as well as reduce pool temperature. Adding sufficient water to the suppression pool to overflow the weir wall will increase the amount of water in the drywell and the reactor cavity.

Natural deposition will remove airborne fission products from the containment atmosphere. The fission products released to the environment will be significantly reduced if the time of FP release is delayed. Airborne fission products can also be removed from the containment atmosphere by the operation of containment sprays. The fission product removal function of the containment sprays is most desirable if there is a drywell failure and the SP FP scrubbing capability is lost. The operation of containment sprays should be carefully monitored because a steam-inerted containment may be de-inerted by containment sprays, and detrimental hydrogen burns may be initiated.

After containment failure, containment pressure provides the driving force for FP release. The strategies that can be used to reduce the containment pressure discussed in the previous section can be used here to reduce the driving force for fission product release.

The release of FP after CF can be either directly to the environment or through the secondary containment. The rate of FP release depends on the pressure in the containment and the size and location of the failure. The FP release can be reduced if the leak area can be identified and flooded. The flow from the containment atmosphere will then pass through a pool of water where some of the fission products will be retained. Analytical results on containment performance such as those presented in NUREG-1037 can be used to provide information about possible leak sites and ways to flood these areas. For FPs that are released through the secondary containment, the standby gas treatment system (SGTS) in the secondary containment can be used to reduce FP release to the environment by HEPA and charcoal filters. The SGTS operation can significantly reduce the release of non-noble gas fission products to the environment if the flow from the primary containment to the secondary containment is within the capacity of the SGTS. For cases where the flow rate exceeds the SGTS capacity (e.g., due to high containment pressure and/or a large leak area), the pressure in the secondary containment will increase, and leakage directly to the environment, as well as failure of the enclosure building (upper part of the secondary containment for Grand Gulf) may result. In the Grand Gulf NUREG-1150 analyses, no credit is given for fission product retention in the secondary containment. A substantial portion of the enclosure building is expected to fail at containment failure. Even under conditions where substantial leakage from the secondary containment develops, the operation of the SGTS will still be beneficial because part of the leak flow will pass through the filters of the SGTS. If the break area is within the auxiliary building where fire spray is available, the fire spray system can be used to scrub fission products from the secondary containment atmosphere and consequently reduce the release of fission products to the environment.

Besides containment failure, release of fission products to the environment will also occur if there is an isolation failure or if an interfacing systems LOCA occurs. These are low probability, but high consequence, events. A containment isolation failure, although not likely in a Mark III containment, may result in the leakage of radioactive material to the secondary containment or directly to the environment. The BWR EPGs have provided guidance to identify and isolate such leaks. In cases where the failed system cannot be identified and isolated, the result will be similar to that of any other containment failure and the strategies discussed above can be applied.

Interfacing systems LOCA (ISL) is a very unlikely, but high consequence, event. In an ISL, the radioactive material in the RCS can escape directly to the secondary containment or the environment, bypassing the containment. This occurs when a failure of the pressure isolation valves (PIVs) between the high pressure and low pressure systems results in the rupture of the low pressure piping from excessive pressure. Should a containment bypass occur, the release could be reduced by reducing the RCS pressure, i.e. the driving force for the release. The release could also be reduced by flooding the pipe that leads to the leak area or keeping the leak area submerged under water, both of which are practical only when the RCS pressure is low. A flooded or submerged break would result in the trapping of some fission products in the water and thus reduce the amount of release to the environment. Finally, if the system that contains the break could be isolated the release would be stopped.

4.3 The Safety Objective Tree

The results of the above strategy identification effort are summarized in the safety objective tree shown in Figure 4.6. As indicated in Section 2.2, for containment and release management, two principal safety objectives exist: maintaining containment integrity and mitigating fission product releases to the environment. 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 venting strategies intended to prevent uncontrolled containment failure, it becomes important to minimize the amount of fission products released under these circumstances. Figure 4.6 was constructed according to the process defined in Section 2.2 and the results of strategy identification presented in Tables 4.3 through 4.6. It systematically defines the challenges to the overall safety objectives for a Mark III containment, identifies safety functions that need to be preserved to meet the objective and lists the specific challenges found in a Mark III containment during a severe accident which could interfere with maintaining these safety functions. Various mechanisms which could cause the challenges are listed and strategies which may be able to prevent the mechanisms from occurring, or which can mitigate their effect, are identified.

As can be seen from Figure 4.6, a particular strategy is often used for many different mechanisms and their associated challenges. This indicates that the same or very similar actions may be taken for a variety of reasons and that once such an action is taken it can have a beneficial effect on arresting and mitigating a number of mechanisms besides the ones which may have originally triggered its implementation. This point is further developed in the detailed strategy description presented in Section 5.

Table 4.1 PDS Core Damage Frequencies for Grand Gulf (Reference 7)

PDS Number	PDS Name	MCDF (1/yr)	PDS % TMCDF
1	Fast Blackout(1)	3.2E-06	79.2
2	Fast Blackout(2)	4.6E-08	1.1
3	Fast Blackout(3)	1.5E-07	3.7
4	Slow Blackout(4)	3.7E-08	0.9
5	Slow Blackout(5)	2.3E-09	<< 1
6	Slow Blackout(5)	1.4E-09	<< 1
7	Fast Blackout(6)	4.2E-07	10.3
8	Slow Blackout(6)	6.3E-08	1.5
9	Fast ATWS	5.0E-08	1.2
10	Slow ATWS	6.2E-08	1.5
11	Fast T2	1.8E-08	0.4
12	Slow T2	2.9E-10	<< 1

Note:

(1) In PSD1, core damage occurs in the short term and the RPV is at high pressure. Offsite power may be recovered. The following functions are available after power recovery: coolant injection, heat removal by containment spray, and miscellaneous systems (venting, SGTS, containment isolation, and HIS).

(2) PDS2 is similar to PDS1, except that heat removal via the sprays is not available after power recovery.

(3) PDS3 is similar to PDS2, except that only low pressure injection with condensate is available after power recovery.

(4) PDS4 is similar to PDS1, except that core damage occurs in the long term and the RPV is at low pressure.

(5) The relationship between PDS5 (PDS6) and PDS4 is the same as that between PDS2 (PDS3) and PDS1.

(6) In PDS8 and PDS9, offsite power is not recoverable.

Table 4.2 Typical Timing for Different Accident Phases of Various Accident Sequences (STCP Calculation)

Accident Sequence	Very Early Phase	Early Phase		Late Phase
	Accident Initiation (hr)	Onset of Core Melt (hr)	Vessel Breach (hr)	
TB1[1]	0	9.5	11.5	CF(21)[2]
TB2[3]	0	9.5	11.5 CF(11.5)	
TBS[4]	0	1.5	3 CF (3)	
TQUV[5]	0	1.5	3.5	CF(14)
TC[5]	0 CF(1.5)	2	4	
TPI[7]	0 CF(22)	27	33	

Note:

(1) TB1 is a station blackout sequence with plant batteries depleted six hours after accident initiation. Hydrogen burn is assumed not to occur and CF is due to the buildup of noncondensibles.

(2) CF (21) means containment failure occurs at 21 hours after accident initiation.

(3) TB2 is similar to TB1, except that hydrogen burn occurs at VB and fails the containment.

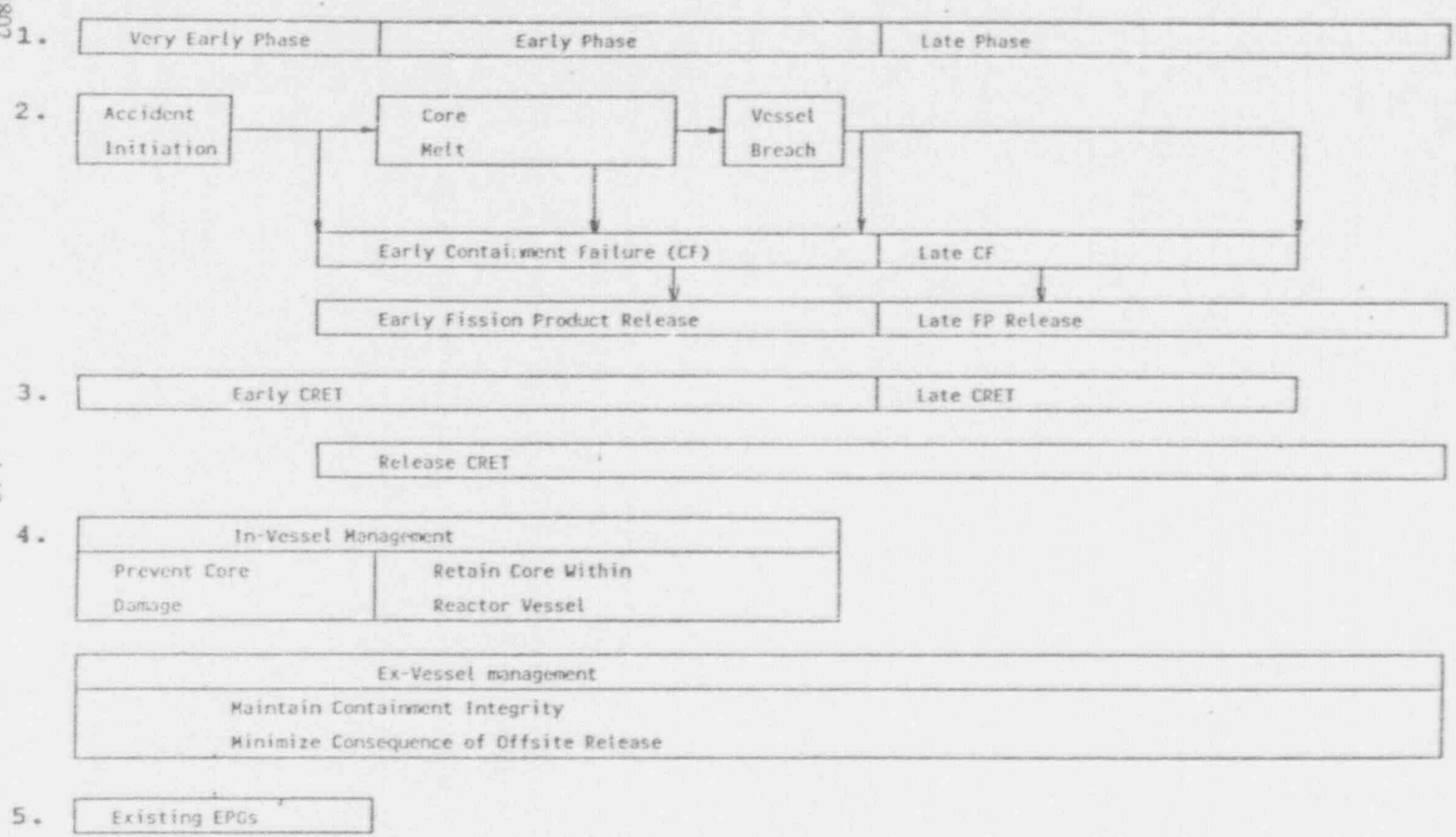
(4) TBS is a SBO sequence with the loss of all makeup water at the beginning of the accident. Hydrogen burn occurs at VB and fails the containment.

(5) TQUV is a transient with the loss of all makeup water (similar to TBS). However, hydrogen igniter system is operable.

(6) TC is an ATWS sequence. CF is assumed to cause the loss of all makeup water.

(7) TPI is a transient with stuck open relief valve (SORV) and loss of containment heat removal (CHR).

Figure 4.1 Time Phases of Accident Progression and Accident Management



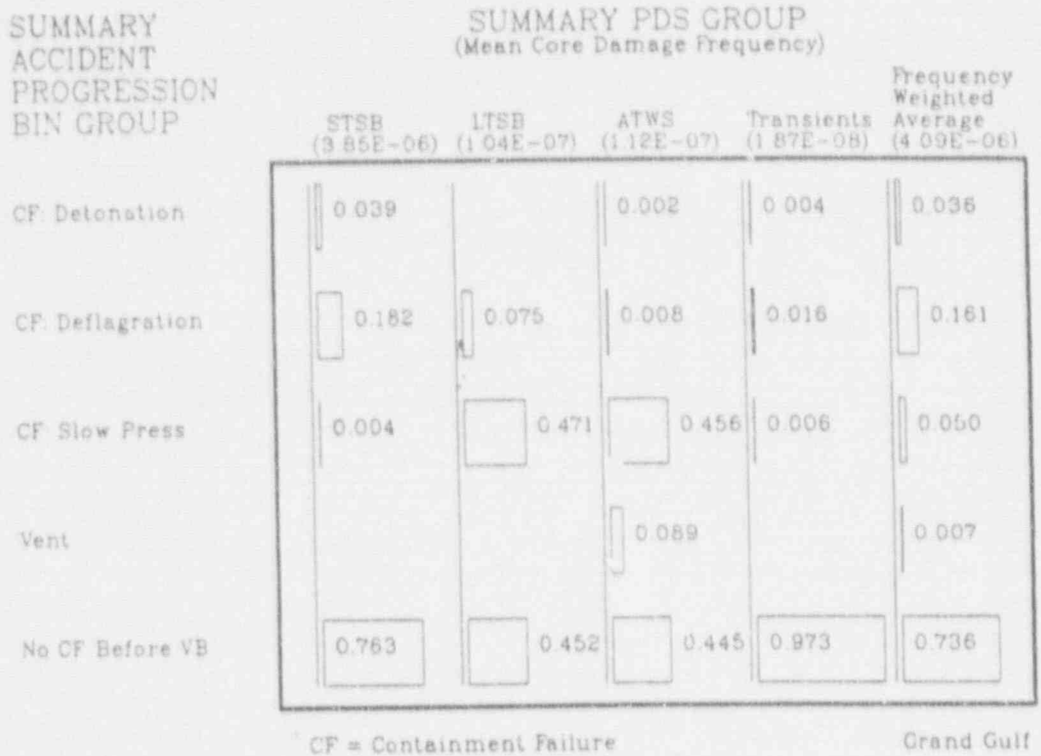
Notes:

1. Time Phases of Accident Progression
2. Phenomenological Events
3. Containment and Release Event Trees (CRETs)
4. Severe Accident Management Activities (SECY-88-147)
5. Applicability of Existing EPGs

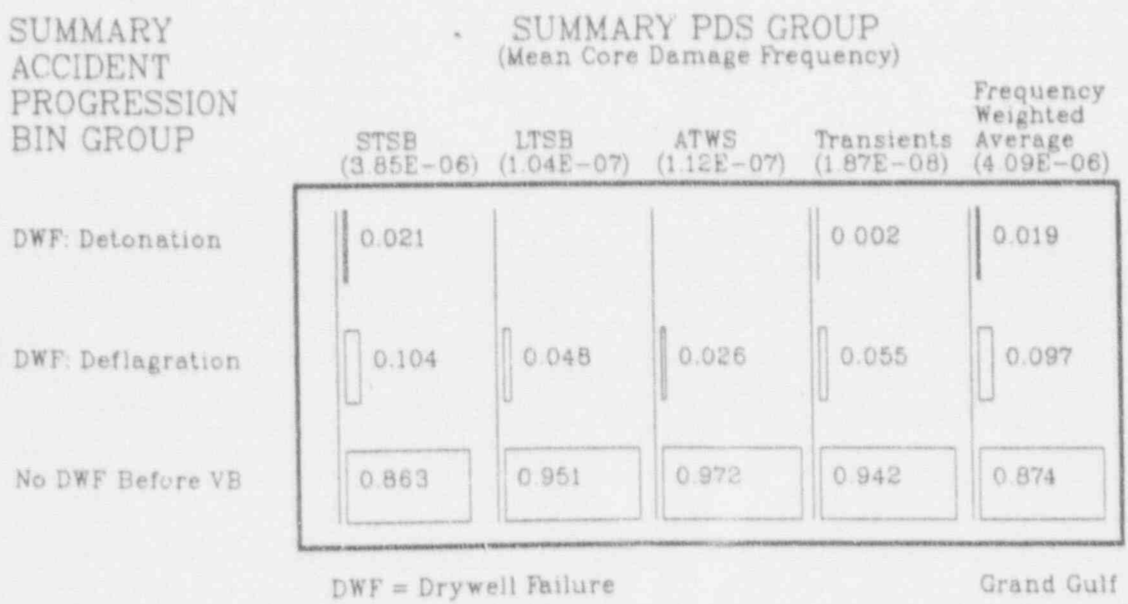
APB NO.	ACCIDENT PROGRESSION BIN	STSB (3.85E-06)	LTSB (1.04E-07)	ATWS (1.12E-07)	Transients (1.87E-08)	Frequency Weighted Average (4.09E-06)
		PLANT DAMAGE STATE (Mean Core Damage Frequency)				
1	VB, early CF, early SPB, no CS	0.16	0.29	0.006	0.011	0.15
2	VB, early CF, early SPB, CS	0.031	0.017	0.23	0.20	0.049
3	VB, early CF, late SPB	0.006	0.005	0.003	0.003	0.007
4	VB, early CF, no SPB	0.18	0.53	0.50	0.33	0.21
5	VB, late CF	0.30	0.12	0.074	0.23	0.28
6	VB, venting	0.032	0.003	0.109	0.075	0.038
7	VB, No CF	0.053	0.003	0.036	0.092	0.050
8	No VB	0.20	0.015	0.025	0.050	0.18

Key:
CF = Containment Failure
CS = Containment Sprays
CV = Containment Venting
SPB = Suppression Pool Bypass
VB = Vessel Breach

Figure 4.2 Conditional Probability of Accident Progression Bins at Grand Gulf (Reference 4)



a. Containment



b. Drywell

Figure 4.3 Mean Probability of Failure Before Vessel Breach (Reference 7)

SUMMARY ACCIDENT PROGRESSION BIN GROUP	STSB (3.85E-06)	LTSB (1.04E-07)	ATWS (1.12E-07)	Transients (1.87E-08)	Frequency Weighted Average (4.09E-06)
DWF: Loads Accompanying VB	0.084	0.158	0.156	0.146	0.088
DWF: Pedestal Fail. (Loads Accomp. VB)	0.051	0.090	0.109	0.096	0.058
DWF: Pedestal Fail. (Dynamic Loads)	0.019	0.018	0.007	0.008	0.016
DWF: Detonation	0.017		0.003	0.008	0.016
DWF: Deflagration	0.022	0.011			0.019
Alpha	0.005	0.006	0.002	0.002	0.005
DWF: Before VB	0.106	0.043	0.026	0.055	0.099
No Early DWF	0.686	0.672	0.696	0.683	0.688

SUMMARY PDS GROUP (Mean Core Damage Frequency)

DWF = Drywell Failure

Grand Gulf

a. Drywell

SUMMARY ACCIDENT PROGRESSION BIN GROUP	STSB (3.85E-06)	LTSB (1.04E-07)	ATWS (1.12E-07)	Transients (1.87E-08)	Frequency Weighted Average (4.09E-06)
CF: Detonation	0.032		0.006	0.016	0.030
CF: Deflagration	0.188	0.305	0.288	0.521	0.208
Alpha	0.003	0.004	0.001	0.003	0.003
CF Before VB	0.219	0.544	0.552	0.022	0.248
No Early CF	0.545	0.145	0.151	0.437	0.499

SUMMARY PDS GROUP (Mean Core Damage Frequency)

CF = Containment Failure

Grand Gulf

b. Containment

Figure 4.4 Mean Probability of Failure at Vessel Breach (Reference 7)

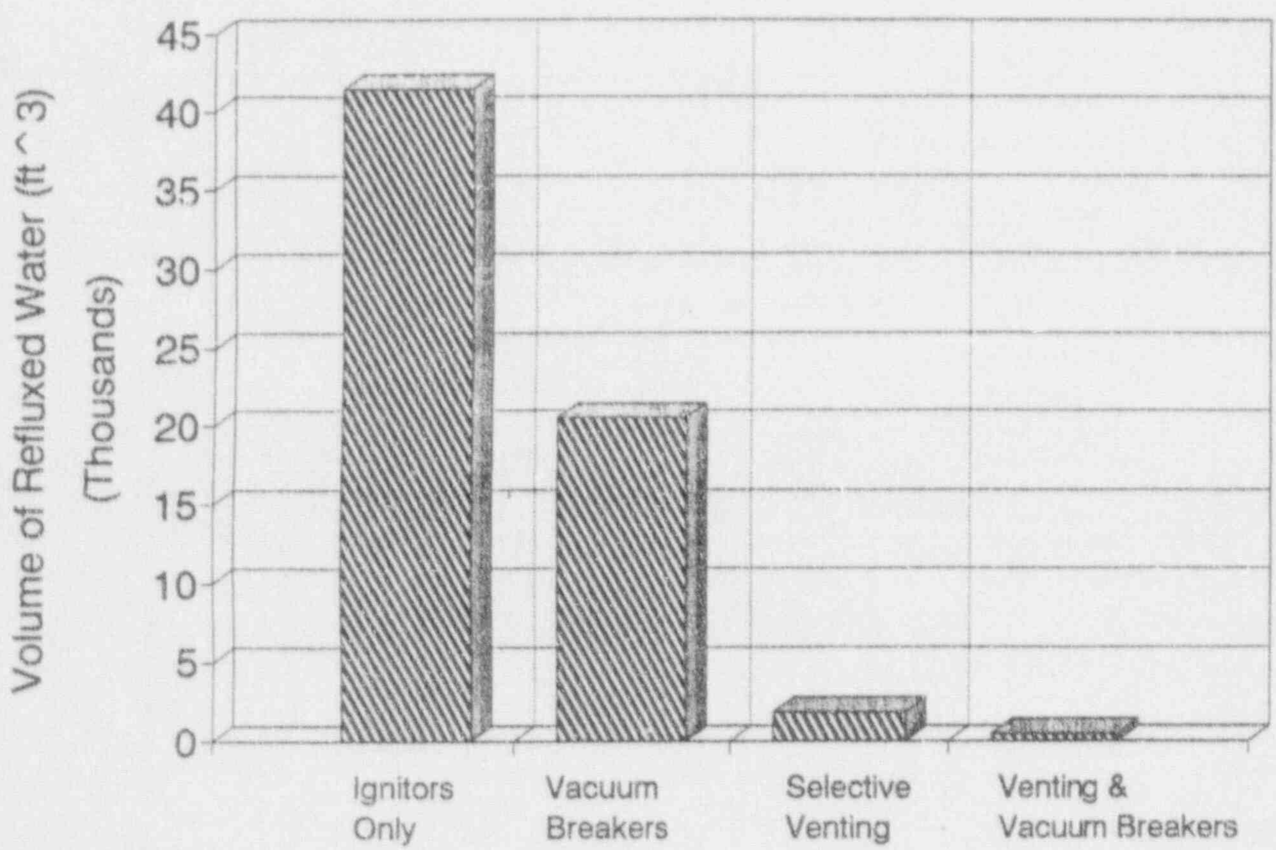


Figure 4.5 Impact of Mitigative Actions on Drywell Cavity Water Inventories for Mark III Short-Term Station Blackout (Reference 13)

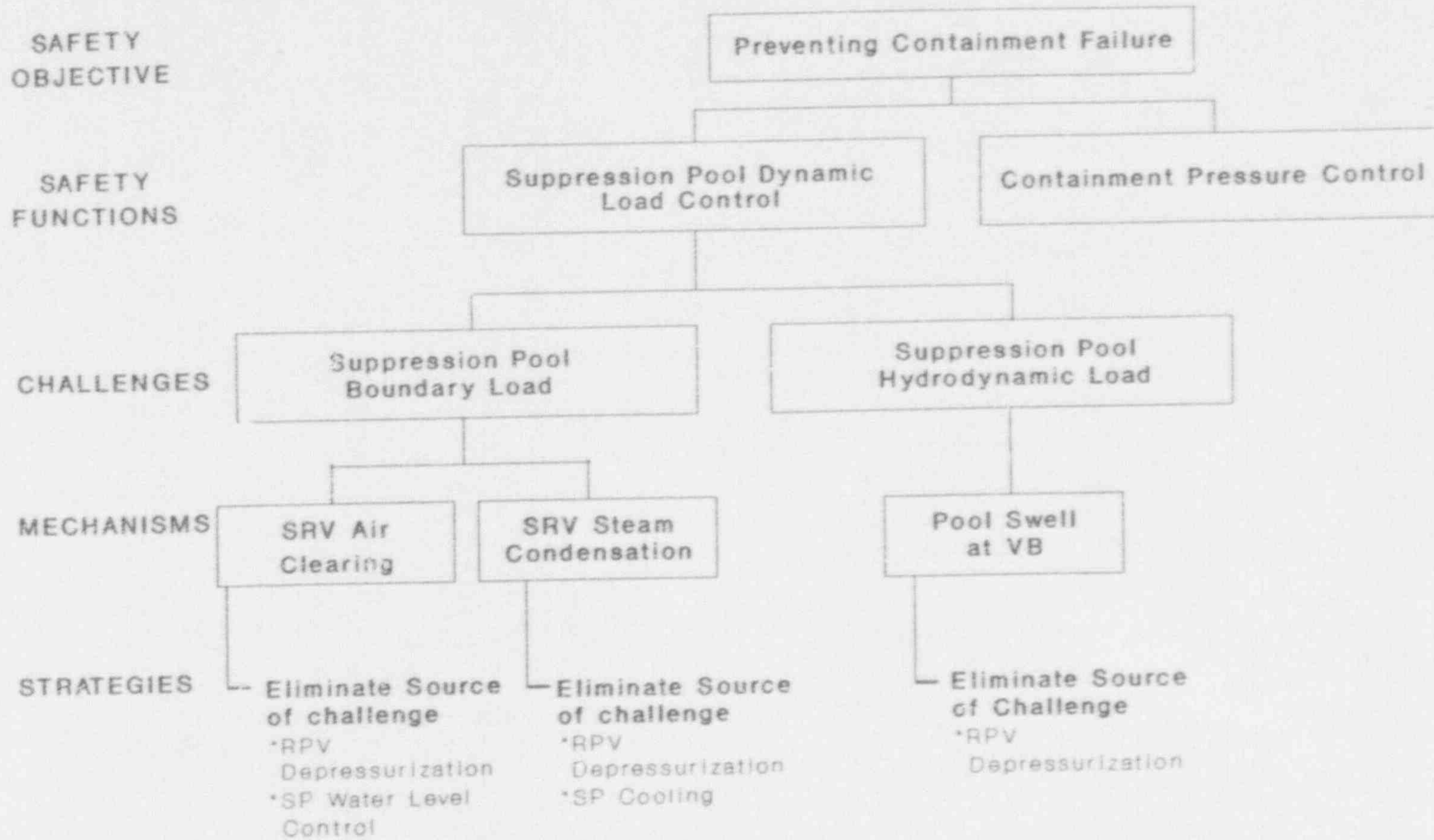


Figure 4.6 Safety/Objective Tree With Identified Strategies for a Mark III Containment (Continued)

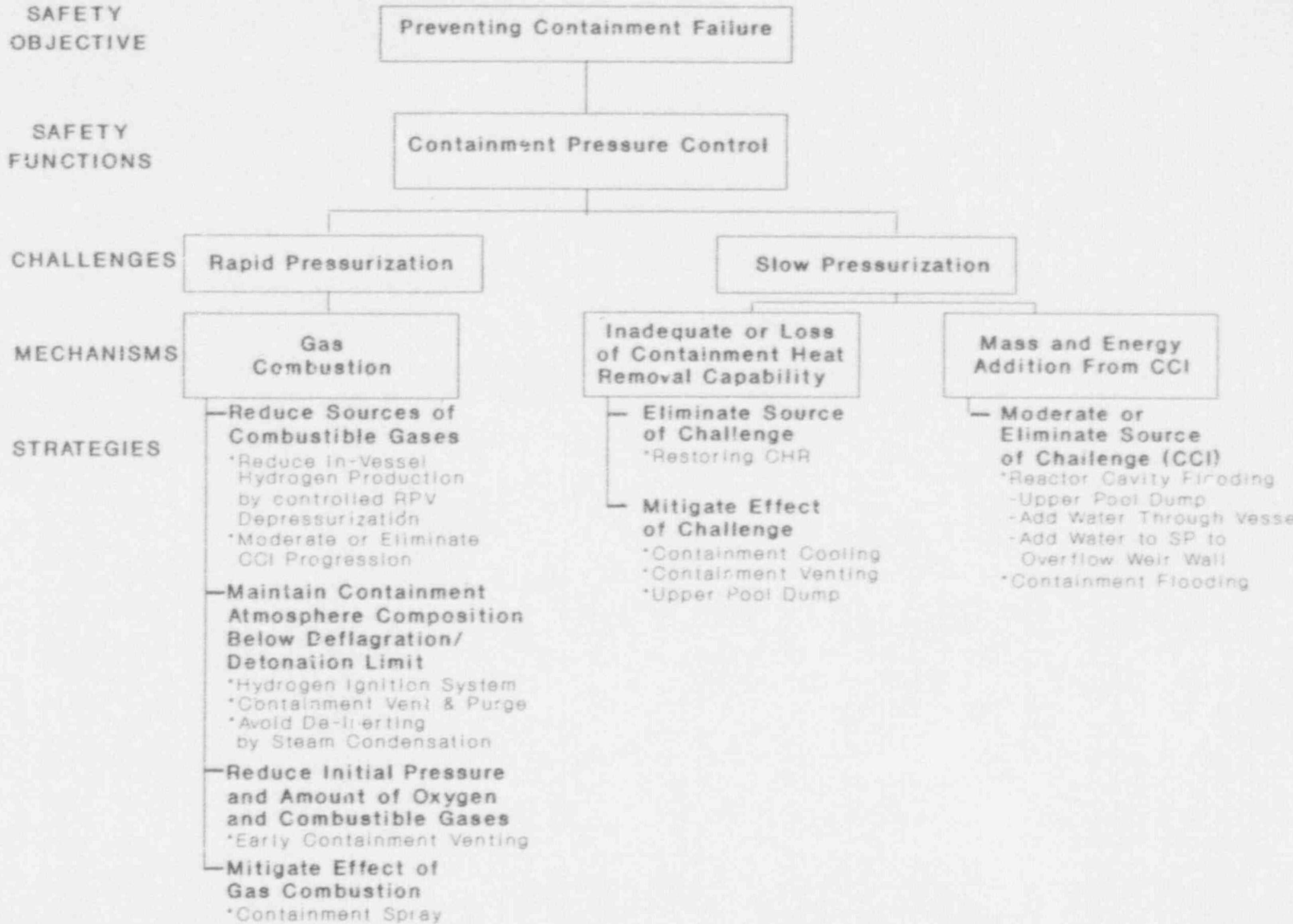


Figure 4.6 Safety/Objective Tree With Identified Strategies for a Mark III Containment (Continued)

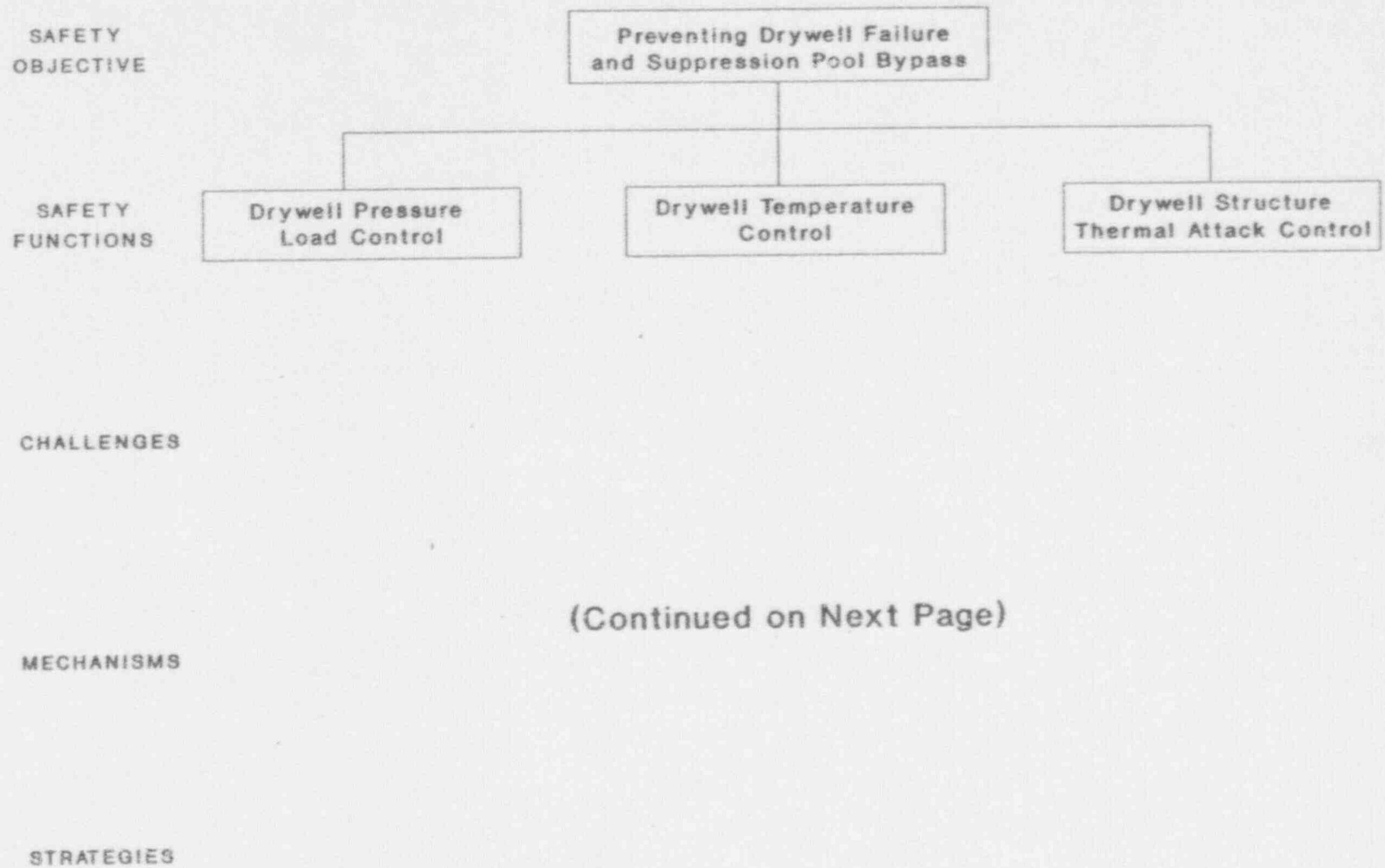


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Figure 4.6 Safety/Objective Tree With Identified Stratagies for a Mark III Containment (Continued)

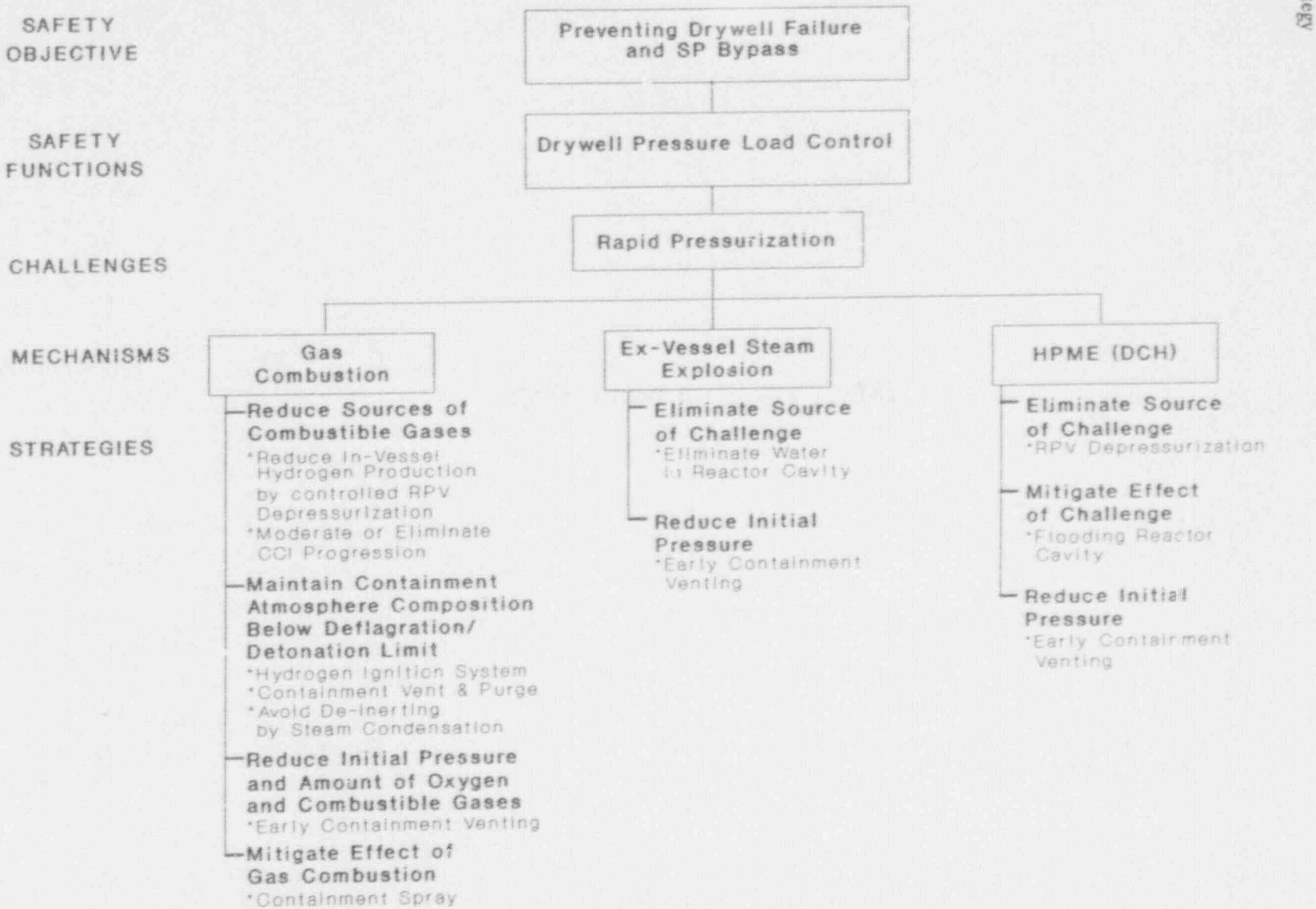

Figure 4.6 Safety/Objective Tree With Identified Strategies for a Mark III Containment (Continued)

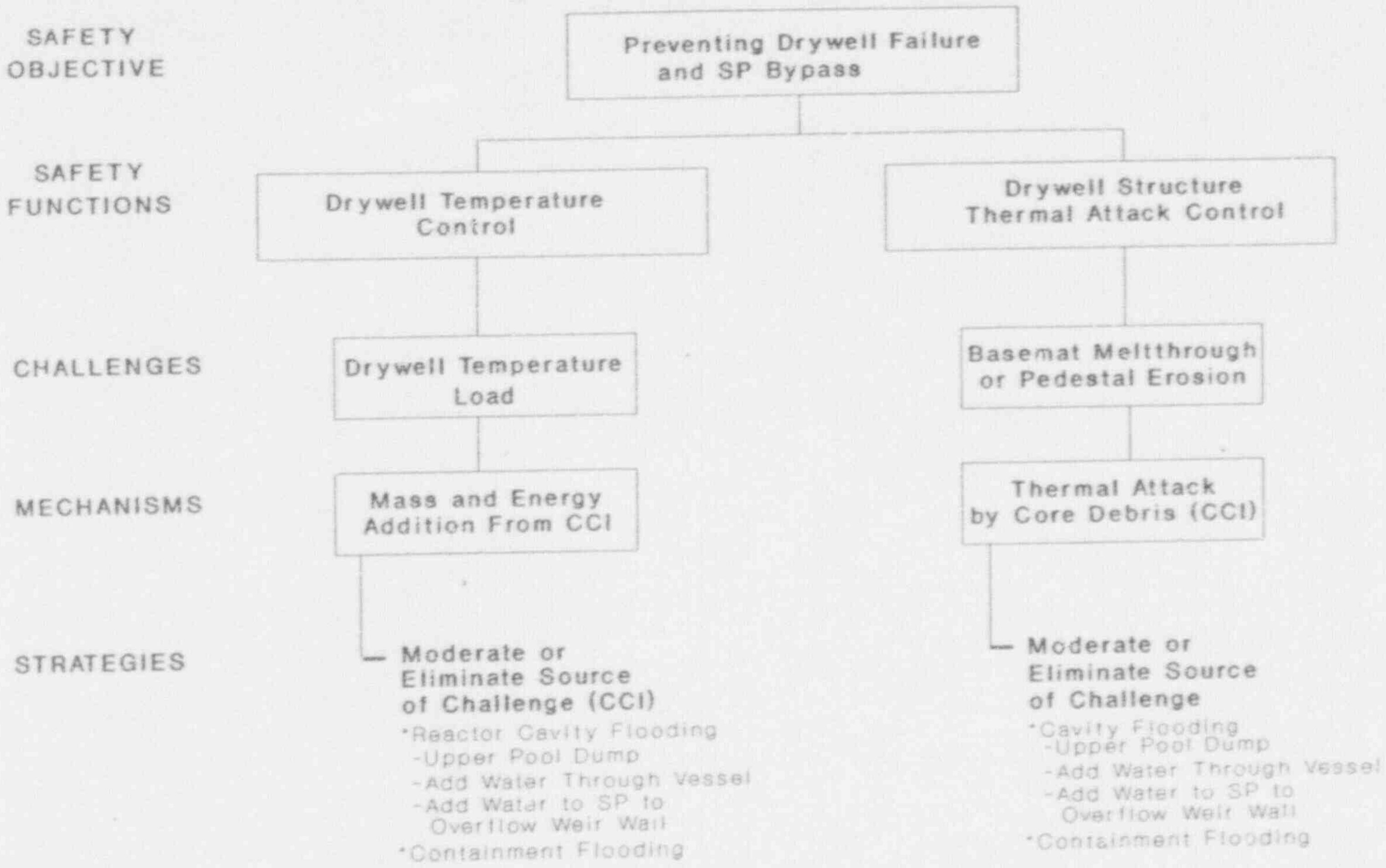

Figure 4.6 Safety/Objective Tree With Identified Strategies for a Mark III Containment (Continued)

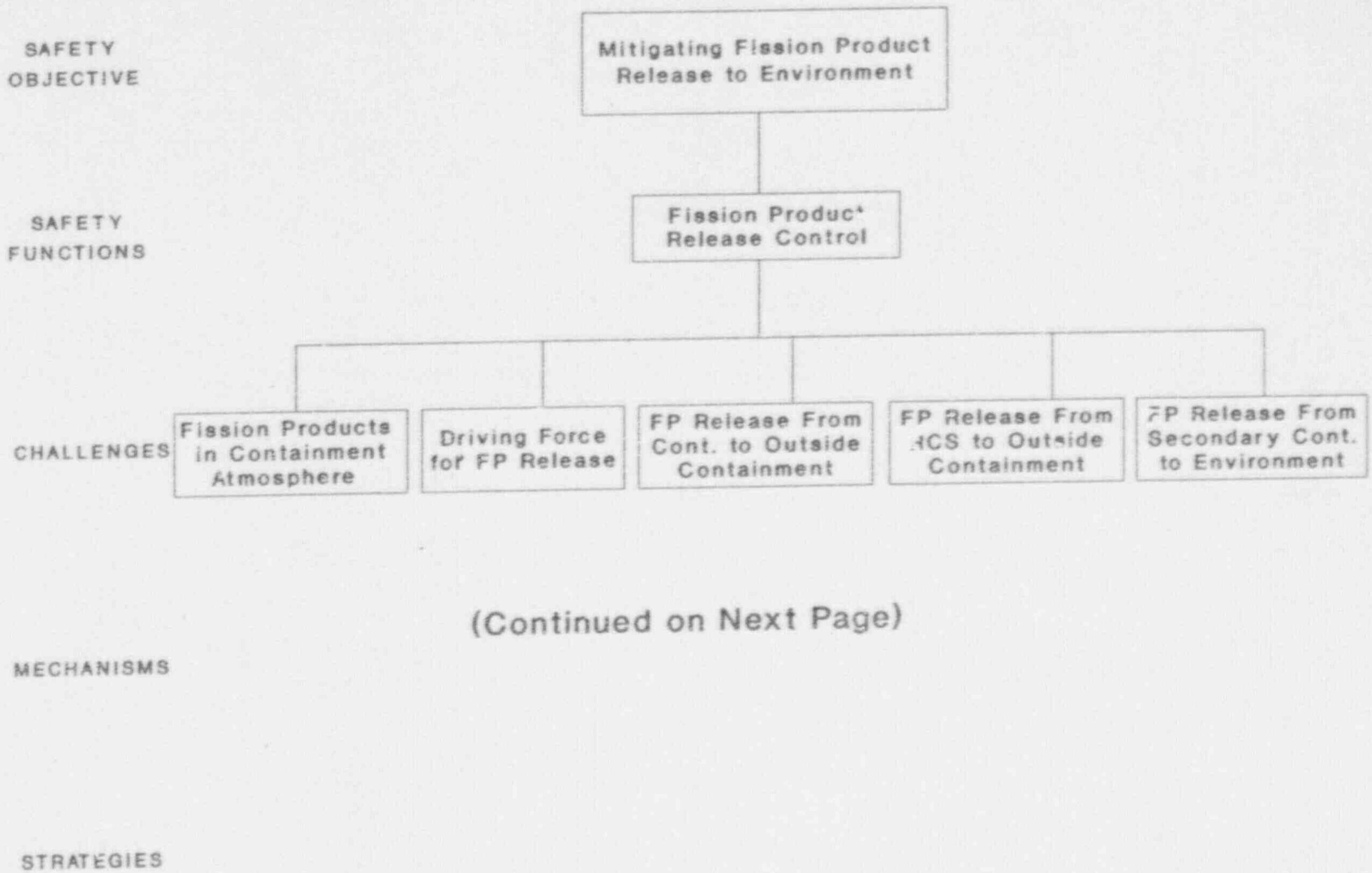

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Figure 4.6 Safety/Objective Tree With Identified Strategies for a Mark III Containment (Continued)

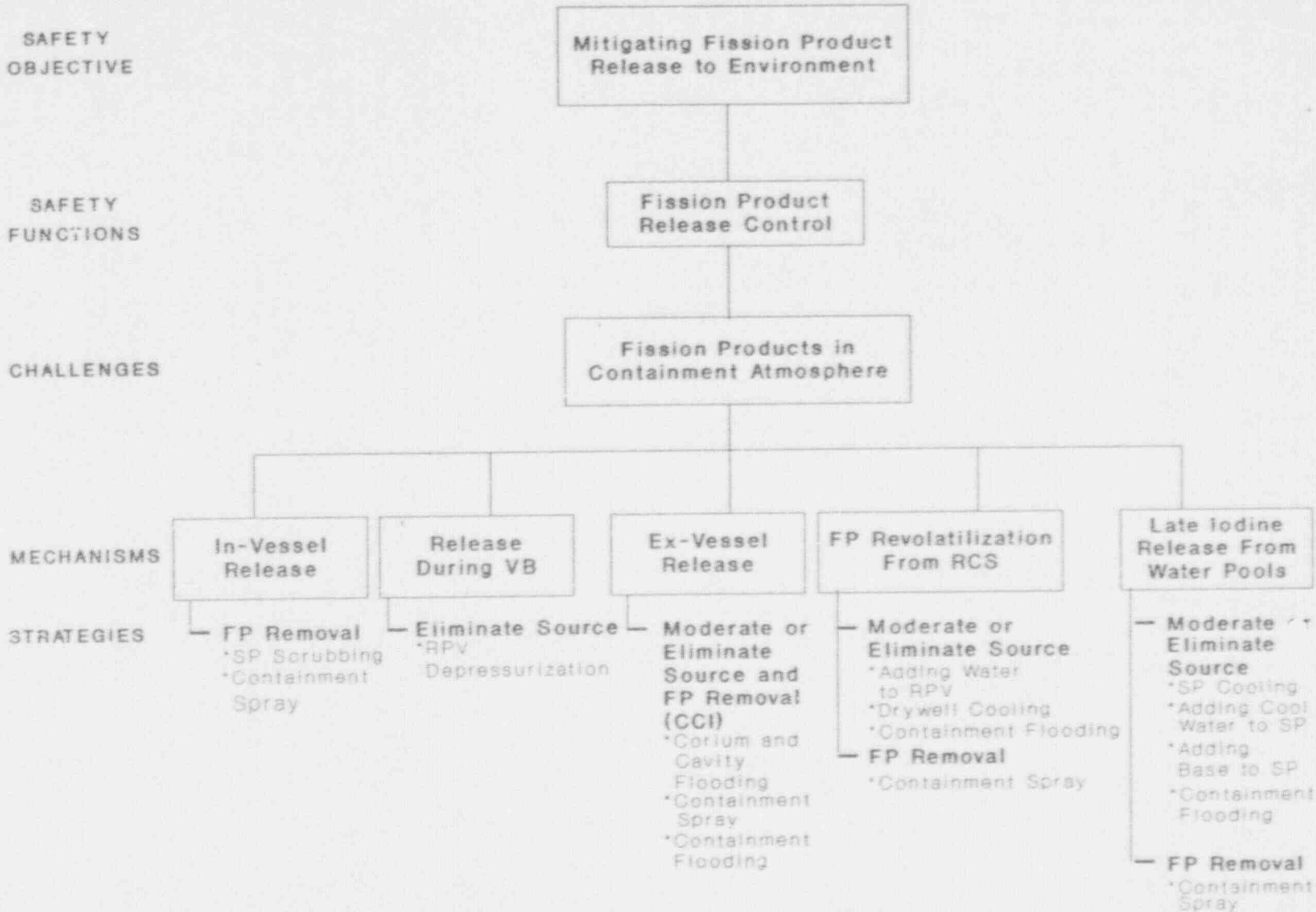

Figure 4.6 Safety/Objective Tree With Identified Strategies for a Mark III Containment (Continued)

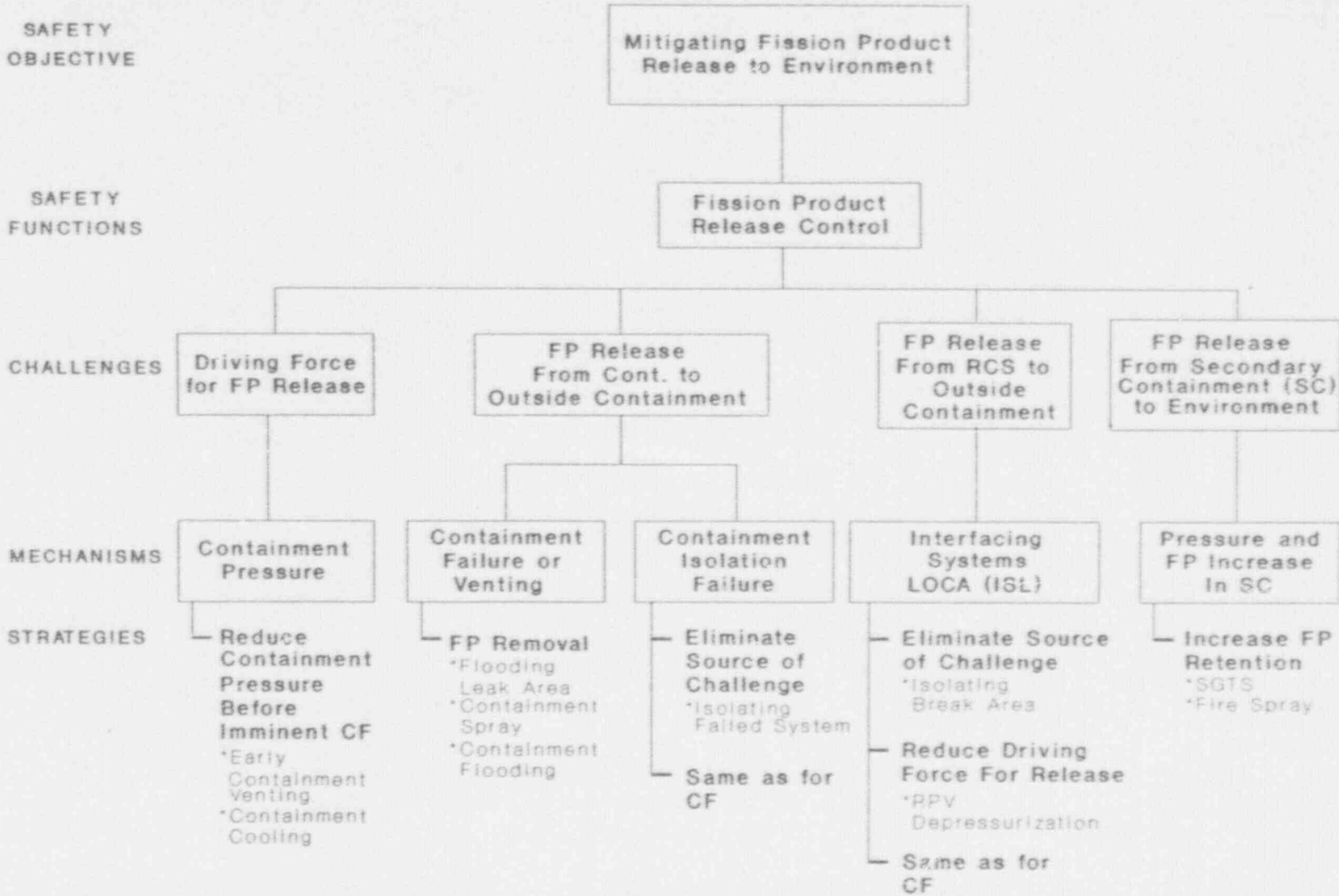

Figure 4.6 Safety/Objective Tree With Identified Strategies for a Mark III Containment (Continued)

5 Strategy Discussion

This section provides a detailed description of the strategies identified in the previous sections. The challenges that can be arrested or mitigated by these strategies and the parameters that can be used to identify these challenges are also discussed.

5.1 Strategies and the Challenges Addressed by the Strategies

Table 5.1 lists the challenges identified in the previous sections and the parameters that can be used to identify these challenges. Actions, or strategies, would be implemented when certain predetermined conditions are reached. For some challenges, direct instrument indication is available, while for others indirect parameters must be used to infer the existence of the challenge.

The instruments that can be used to obtain the important parameter values during and following an accident are described in Regulatory guide 1.97 (Rev. 3) [28]. Control room instrumentation information is also provided by the Safety Parameter Display System (SPDS), which is required by the NRC as part of a nuclear plant's emergency response capability [25]. Two important issues determine the availability of instruments during a severe accident. The first is their survival under severe accident conditions and the second is their availability during a station blackout.

The environmental qualification of the plant instruments include consideration of temperature, pressure, humidity, and radiation conditions. Typical containment instrument qualification pressure and temperature, as required by Regulatory Guide 1.89 and IEEE 323-1974, are approximately 85 psia and 350°F, respectively [30]. The actual environmental conditions in a severe accident may be considerably harsher, particularly the temperature in the drywell, if a corium concrete interaction (CCI) has been in progress for some time.

The availability of instruments during station blackout (SBO) sequences is important because station blackout contributes significantly to the total core damage frequency for Grand Gulf (and most likely for other Mark III plants also). Lack of instrument indication during SBO presents a serious problem for CRM, particularly after the depletion of plant batteries. There is no specific requirement for an independent power supply for containment instruments. Identification of the instruments that are available and reliable during a station blackout, or after depletion of station batteries, is therefore important. The identification of other methods to obtain essential parameters under SBO conditions is also important. A more detailed discussion of this issue has been presented in the Mark I report [32].

Table 5.2 correlates the strategies identified in Section 4 with the challenges presented in Table 5.1. Table 5.2 shows that most of the strategies have the potential of addressing a variety of challenges, and once implemented they may have many beneficial effects. On the other hand, some strategies while beneficial for some of the challenges, may aggravate or precipitate other challenges.

5.2 Strategy Description and Discussion

The strategies presented in Table 5.2 are described in more detail in this section. The information discussed in the previous sections is integrated to provide guidance for the development of CRM strategies which could be considered for implementation.

5.2.1 Strategies Related to Resource Management

The implementation of the CRM strategies listed in Table 5.2 requires plant systems such as RHR or SSW, and resources such as electric power, pneumatic supply, and water. Section 3 provided a detailed discussion of the plant systems and resources that can be used for CRM.

One of the most important water sources for plant safety systems is the suppression pool (SP). During a severe accident, the SP temperature may become high enough to cause accelerated pump wear, or the water level may become low enough to prevent the pool from being a viable water source. It is then necessary to switch to a cool alternate water source. Additional discussion of this topic can be found in Section 3.1.2.

The electric power and pneumatic supply for Grand Gulf have been discussed in section 3.2.2, along with the strategies to extend the availability of electric power or to enable emergency replenishment of the pneumatic supply.

5.2.2 Strategy to Depressurize the RPV

RPV depressurization is one of the key actions contained in the BWR EPGs. Emergency RPV depressurization is called for under the primary containment control guideline when (1) the SP temperature cannot be maintained below Heat Capacity Temperature Limit (HCTL, about 150°F for RPV at system pressure for Grand Gulf), (2) the drywell and containment temperature cannot be maintained below their design temperature limits (330°F for the drywell and 185°F for the containment for Grand Gulf) (3) the containment pressure cannot be maintained below the pressure suppression pressure (PSP, 9 psig at normal SP water level for Grand Gulf), or (4) the SP water level cannot be maintained above the heat capacity level limit (HCLL) or below the SRV tail pipe level limit (TPLL, for SRV air clearing load consideration). RPV depressurization is also called for in the RPV control guideline of the EPGs and may occur automatically when some plant conditions are reached, e.g., a low RPV water level and a high drywell pressure condition.

The RPV will most likely be depressurized during the course of a severe accident. However, due to loss of electric power, loss of pneumatic supply or insufficient supply pressure, or operator error, the system may not remain depressurized. Since depressurization requires dc power, the RPV will be pressurized again in a station blackout sequence after battery depletion.

As a result of the CPI program, the NRC staff has recommended to the commissioners an enhanced RPV depressurization system for Mark I containments [36]. This Mark I improvement is also recommended for consideration for the Mark III containments [37]. The recommended ADS enhancements for Mark I containments include the assurance of electric power beyond the requirements of existing regulations, improvement in the temperature capability of the cables (from 340°F to 800 or 1600°F), an additional nitrogen bottle for each ADS valve to allow longer operation (up to 16 hours), and a logic change to provide more complete automation for ISL events. This enhanced RPV depressurization reliability would significantly reduce the likelihood of high pressure scenarios such as those from station blackout sequences.

As a CRM strategy, RPV depressurization before substantial core damage has developed could help (1) to avoid SP boundary loads when a significant amount of noncondensible gases is generated in the RPV from cladding oxidation, (2) to avoid the challenges associated with HPME, and (3) to reduce the amount of FPs released to outside the containment during an ISL event. Unlike the Mark I situation, the loads associated with HPME in a Mark III containment do not challenge the containment directly. Instead, the HPME loads in a Mark III containment challenge the structural integrity of the drywell. A drywell failure in a Mark III containment will result in a suppression pool bypass and additional pressure load and fission product release to the containment.

RPV depressurization may also reduce the amount of in-vessel FP release (release before VB) to the containment atmosphere because of the greater FP decontamination factor of the SRV sparger than that of the SP vents. A carefully controlled RPV depressurization (at an optimum RPV water level) may also reduce the amount of in-vessel hydrogen production and thus the challenge of early containment and/or drywell failure by hydrogen combustion.

RPV depressurization also allows a gradual release of the hydrogen produced in the vessel and benign hydrogen burns in the containment if the HIS is operating. Without RPV depressurization, significant amounts of hydrogen may be accumulated in the vessel before VB. The release of this hydrogen may cause a severe hydrogen burn to occur in the containment at VB.

The parameters that can be used to identify these challenges are shown in Table 5.1. RPV in-vessel instrument indications, e.g., core temperature, are required to estimate the potential for or degree of cladding oxidation, the corresponding amount of hydrogen generated, and the probability and timing of vessel breach. An ISL event would be indicated by a high temperature and radiation level outside the containment and a relatively low temperature and radiation level in the primary containment.

Early RPV depressurization may accelerate the in-vessel core melt progression after the loss of core injection, and shorten the time to vessel breach. RPV depressurization may also increase the probability of an Alpha mode failure. The beneficial effects of RPV depressurization are in general more important, particularly after significant core damage has developed and core melt continues.

5.2.3 Strategies Related to Containment Venting

Containment venting is recommended in the BWR EPGs as a means to prevent containment failure due to high pressure. The BWR EPGs provides EOP guidance for the operator to carry out containment venting before the pressure reaches the primary containment pressure limit (PCPL). As indicated in Table 5.2, venting can also be useful for other reasons. These uses are: (1) to prevent contai[illegible]ent pressure failure by reducing the base pressure before mechanisms that may cause rapid pressurizati[illegible]ke effect, and (2) to reduce the wetwell pressure and thus the potential of reactor cavity flooding bef[illegible]3. The adverse effects associated with venting, as shown in Table 5.2, are (1) the release of FP to the environ[illegible]nt, and (2) the drywell temperature load during CCI [32]. These issues of containment venting are discussed i[illegible] following.

<u>Containment Venting to Prevent Containment Pressure Failure:</u> Containment venting has been described as a "last resort" effort to prevent containment failure and uncontrolled fission product release to the environment. To avoid exceeding the PCPL, the BWR EPGs call for venting even if the permitted offsite radioactivity release level is exceeded.

Containment venting is the only action that plant personnel can take to prevent a containment pressure failure due to noncondensible gas buildup. The containment venting systems were not originally designed for severe accident conditions. Therefore, some important issues, e.g., the flow capacity of the selected vent paths, their structural capability, and their operability under severe accident conditions, should be investigated when establishing a containment venting program. These issues have been discussed in detail in the Mark I report.

The determination of the venting pressure, PCPL, is another important issue. Starting venting at too low a value may cause unnecessary release of fission products to the environment while a higher value increases the potential for containment failure. The containment venting pressure for Grand Gulf is 17.25 psig, which is far lower than the expected containment failure pressure of about 56 psig.

Presently there is no guideline in the BWR EPGs on when to reclose the vent path(s). It would be desirable that guidelines, based on pressure and vent path operational considerations, be provided for vent reclosing to minimize the release of fission products. Such a requirement is provided in Grand Gulf plant specific EOPs. The Grand gulf EOPs require the operator to close the vent path at a pressure of 17 psig. The pressure difference between opening and closing the vent valves is only 0.25 psi. A small pressure difference may result in many opening-closing operations, and thus an increase in the failure probability, of the vent valves.

Direct instrument indication is available for containment pressure. The post-accident primary containment pressure measuring system covers a range from -5 psig to 3 times design pressure for concrete and 4 times design pressure for steel containments [28]. Since the PCPL is usually taken to be one to two times the design pressure, this range is sufficient. However, the pressure indication may not be available after the loss of electric power, and this presents a serious problem for containment venting in SBO sequences because PCPL is most likely reached after the depletion of plant batteries.

Early Containment Venting: Early containment venting, i.e., containment venting at a pressure lower than the PCPL, can be used to reduce the initial containment pressure, in anticipation of a sudden, large pressure increase associated with a HPME, EVSE, or hydrogen burns, and thus prevent catastrophic early containment and/or drywell failure. However, early containment venting may not be very effective in preventing containment and/or drywell failure from these challenges because of the large quasi-static loads associated with these challenges and the low PCPL for the Mark III containments. Besides, it is difficult to estimate the time when these challenges may occur. Nonetheless, early containment venting before significant core damage has occurred may be desirable under certain conditions. The containment atmosphere is relatively clean at this time and the amount of fission products released to the environment will be small.

Containment venting can also be used for combustible gas control. Although containment venting cannot change containment atmosphere composition and thus the challenge of combustion, containment venting, when the containment is steam inerted, can remove combustible gases and oxygen from the containment. Later combustion, after steam condensation, may be averted due to oxygen deficiency. Even if later combustion does occur, the amount of combustible gases available for combustion is reduced.

There are potential adverse effects associated with venting a steam rich containment atmosphere. Unacceptable negative containment pressure loads and/or re-introduction of oxygen to the containment may occur after steam condensation if containment noncondensible gas level has been reduced too low by containment venting. This adverse effect can be avoided by closing the vent path when a sufficient amount of noncondensible gases (e.g., hydrogen) remains in the containment, or if there is a sufficient amount of noncondensible gases generated in the containment after vent closure. Analyses of a station blackout sequence show that after 10 hours the largest contribution to containment pressure rise is due to the CO_2 generated from CCI [13]. The containment can therefore be maintained pressurized, by the production of noncondensible gases, and inerted, due to lack of oxygen, after containment venting.

Containment venting has been shown to be the most effective method to prevent SP water overflow to the drywell and the reactor cavity. This aspect of containment venting will be discussed later in this section when the strategy of eliminating water from the reactor cavity is discussed.

Operator Actions and Equipment Requirements: The operator actions needed to carry out venting strategies include (1) determining that the condition for venting initiation has been reached (2) defeating the containment isolation valve interlocks, and (3) opening the ac motorized valves from the control room. In the case of an SBO event ac power is not available and the operator must open the valves manually wherever the valves are actually located in the plant. However, if the recommendation of the CPI Program has been implemented in a plant, the valves can be operated from the control room by dc power, and this would significantly increase the probability of successful containment venting. The CPI program has also recommended the inclusion of a rupture disk in the vent path. The presence of such a disk will affect the feasibility of using early venting to lower base pressure and containment venting for combustible gas and reactor cavity flooding control. If these strategies are deemed to be important for a particular plant, they must be considered in choosing a disk rupture pressure.

In accordance with the requirements of NUREG-0696 [38], the TSC will provide technical support to the reactor operators. Since venting procedures have been established in the existing EPGs, it is very likely that these procedures will be carried out when the PCPL is reached. However, without explicit guidance, the operator will be reluctant to vent the containment before the PCPL is reached, particularly with the contaminated containment atmosphere which will often exist when venting could be useful. Responsibility for venting decisions should be clearly defined and explicit and unambiguous guidance should be given to the operators.

Potential Adverse Effects: The potential adverse effects discussed in the Mark I report [32] include (1) loss of plant safety equipment due to containment depressurization and SP flashing, (2) secondary containment contamination and resultant loss of function of safety related equipment or loss of accessibility to the secondary containment, and (3) fission product release to the environment. The first two issues listed above are of lesser concern for a Mark III containment because, in a Mark III containment, the ECCS pumps have been designed to pump saturated water, and the piping/duct of the vent path is high in the secondary containment, well above where safety equipment is located. However, accessibility to the secondary containment may still be lost.

The most important adverse effect of containment venting is the release of fission products to the environment. In a Mark III containment, the fission products will be scrubbed by the suppression pool if the drywell is not failed. However, the leakage between the drywell and the containment in a Mark III containment is larger than that in a Mark I containment. This leakage will cause some suppression pool bypass, and its effect needs to be considered. Another adverse effect of containment venting is the increase in drywell temperature if high temperature gases are discharged to the drywell, such as occurs during CCI, and drywell cooling is not available [32]. This may cause an earlier drywell temperature failure in a Mark III containment.

5.2.4 Strategies Related to Containment Spray

As shown in Table 5.2, containment spray, in addition to its designed function of containment pressure and temperature control, also can remove fission products from the containment atmosphere. Furthermore, containment spray, when used with the HIS, may have a potential to mitigate the effects of hydrogen combustion. On the other hand, containment spray, if used when the containment is steam inerted, may condense sufficient amounts of steam and cause the containment atmosphere to become combustible.

The use of containment spray to control containment pressure and temperature under accident conditions is described in the BWR EPGs. Containment spray is called for in the EPGs when the containment temperature reaches the design temperature (185°F for Grand Gulf) or when the containment pressure exceeds the suppression chamber spray initiation pressure (SCSIP, 9 psig for Grand Gulf at normal SP water level). Containment spray is also called for in the BWR EPGs when containment pressure cannot be maintained below PCPL.

The use of the containment spray as a water source for reactor cavity and containment flooding is the topic of the next strategy and will be discussed in Section 5.2.5.

One of the most important functions of containment spray in CRM is its ability to scrub fission products from the containment atmosphere. This function is particularly vital if the discharge of the fission products bypasses the suppression pool, and, as a consequence, the airborne fission product concentrations in the containment are high. The suppression pool will be bypassed in a Mark III containment if the drywell fails.

As a fission product scrubbing tool, containment spray is activated when the radiation level in the containment is high or, if the containment has already been breached, as indicated by the radiation level in the secondary containment or offsite. When operating the containment spray containment pressure and temperature should be constantly monitored to assure that the spray will not lead to a containment failure due to negative containment pressure load (about -3 psid design). The possibility of deinerting the containment atmosphere by steam condensation should also be evaluated. The current BWR EPGs provide guidelines for the derivation of a containment spray initiation limit beyond which containment spray should be prohibited to avoid the unacceptable negative pressure load. Since the spray strategies discussed here can have different objectives and may be implemented under conditions significantly different from those when containment spray is required by the EPGs, different containment spray initiation limits may need to be established [32].

Containment spray for fission product scrubbing is usually required during or after the late phase of an accident, and containment conditions may have exceeded the environmental condition for instrument qualification before this time. Whether there is still sufficient instrument indication available for the management of containment spray is uncertain. Alternate means of obtaining necessary indications may have to be planned in advance.

Once the decision to use the containment spray is made, the operator must line up the RHR system in the containment spray mode, check the emergency procedures to assure that it is safe to operate the spray, and then start the spray. In some cases, such as during SBO, where the normal water supply is not available, the operator must locate and align an alternate water supply that has its own power source, such as the diesel-driven fire water system. The operator must continue to monitor the containment pressure against the containment spray limits during spray operation to assure that spraying will not cause unacceptable adverse effects.

The use of containment spray may result in some potential adverse effects. The primary concerns are the negative pressure load and containment deinerting. The use of an alternate source with untreated water may clog the containment spray system. A decision is also required if only a limited water supply source is available, and the water can be either delivered to the vessel or used for containment spray. Clearly defined procedures or guidelines are needed to avoid confusion in the management of containment spray under these conditions.

5.2.5 Reactor Cavity and Containment Flooding

This strategy involves three parts. The first is flooding the reactor cavity before vessel breach to: (a) provide conditions favorable for cooling the core debris discharged from the RPV, and (b) mitigate the challenges associated with HPME. The second is to continuously add water to the core debris after it falls into the reactor cavity and interacts with concrete, to: (a) moderate or terminate the progress of CCI and (b) provide an overlying water pool for fission product scrubbing. The third is to flood the containment to above the top of the weir wall, or to the top of the active fuel (TAF) level of the reactor core. Flooding the containment to the TAF level (1) may provide water to the core material remaining in the vessel and thus reduce fission product release from revolatilization and (2) will provide water to the corium on the drywell floor and thus terminate, or slow down the rate of, CCI. The water pool will also reduce fission product release by pool scrubbing, and through dilution the large amount of water will also reduce the late release of pool iodine.

Before VB, one action an operator can take to affect the probability and extent of reactor cavity flooding is to dump the upper containment pool. With an upper pool dump, the SP water level will reach almost the top of the weir wall (The design value of the freeboard after upper pool dump is 5 inches for Grand Gulf.) and a small pressure differential between the containment and the drywell can push the SP water over the weir wall to the drywell. The containment pressure rise due to SRV discharge during core degradation may be sufficient to cause such a SP overflow. Hydrogen combustion in the containment will increase containment pressure and thus the amount of SP water overflow. On the other hand, the operation of the drywell-wetwell vacuum breakers and containment venting will reduce containment-to-drywell pressure differential and thus the probability and extent of reactor cavity flooding. In the NUREG-1150 analyses, the probability of a dry reactor cavity before VB is assumed to be low without an upper pool dump and zero with an upper pool dump [7].

A very important adverse effect of reactor cavity flooding before VB is the increase in the probability of a severe ex-vessel steam explosion (EVSE) at VB. A severe EVSE may fail the drywell and result in a suppression pool bypass. The decision on whether early reactor cavity flooding should be promoted or suppressed depends on judgement. Additional discussions of this topic are presented in the next section when the strategy of eliminating water is discussed.

Unless the SP spills over the top of the weir wall, the only means to add water to the reactor cavity after VB is to inject water to the RPV. This can provide a water supply to the reactor cavity for the cooling of the core debris. Continuously adding water either through the RPV or via suppression pool overflow, using water sources external to the containment will flood the primary containment. Primary containment flooding, once achieved, can provide corium cooling and FP scrubbing with minimal use of active systems, and is particularly important if the drywell has failed and the SP is bypassed. Since Contingency #6 of the RPV Control Guideline of the BWR EPGs calls for flooding the containment up to the top of the active fuel (TAF) level of the reactor core, flooding may have been carried out earlier in an accident as an in-vessel strategy to provide core cooling.

Once the primary containment is flooded, the ability to perform other CRM strategies becomes very limited. However, containment flooding may be ve desirable after the vessel is breached because (1) it involves a minimal use of active equipment and (2) it provides the best means to control the progress of CCI and to mitigate the results of CCI, e.g., the cooling of the gases generated during CCI and the scrubbing of the fission products released from CCI. The large amount of water improves the efficiency of pool scrubbing and the retention of the fission products released from both in-vessel and ex-vessel core debris.

The addition of a large quantity of water to the containment will decrease the containment airspace volume and thus the energy absorbing capability of the containment atmosphere and increase the hydrostatic load on the containment. Consequently, the rate of containment pressure rise per unit energy input will be higher. Even if the mass and energy released to the containment atmosphere from CCI is terminated, the energy from the decay heat will raise containment pressure steadily and containment venting may be required to remove the added energy. Therefore, it is important that sufficient containment vent paths are not flooded and remain operational for the duration of the accident. Containment flooding will also cause some plant systems and instrumentation in the containment to be submerged and damaged, and preplanning is necessary to ensure that this would not affect a successful management of the ongoing accident. Furthermore, some instrument taps in the containment may be submerged and their readings affected.

Most of the time, only indirect inferences are available to deduce the existence of the challenges that are addressed by this strategy (Tables 5.1 and 5.2). For example, to assure sufficient water in the reactor cavity before vessel breach, an upper pool dump may need to be initiated considerably before vessel breach. The water level in the drywell may be indicated by the drywell sump level instrument or inferred from some drywell temperature readings (if the temperature sensors are submerged). The drywell sump level instrument normally has a very limited range and may not be working after VB.

As discussed above, upper pool dump is the only means to add water to the reactor cavity before vessel breach. However, after reactor breach, water can be added to the corium either through the vessel by the use of core injection or by containment spray (by overflowing the Weir Wall). Adding water through the vessel can keep the core materials in the vessel cooled and reduce FP revolatilization from the RCS, while adding water via the containment spray can provide an additional fission product scrubbing capability and also cool the containment atmosphere. If drywell integrity is maintained, fission products will be scrubbed and gases will be cooled by the suppression pool, thus keeping containment temperature at acceptable levels. In this case, adding water through the RPV should be preferable. On the other hand, if both containment and drywell have failed and fission products are being released to the environment, containment spray would be preferred.

There are adverse effects associated with reactor cavity flooding. The most significant potential adverse effect is the probability of a severe EVSE failing the drywell. Besides EVSE, an upper pool dump when the hydrogen concentration in the containment is high and the containment is steam inerted may result in sufficient steam condensation to cause containment deinerting and hydrogen burns. Using containment spray to add water to the containment may lead to a similar result. Adding water to the hot corium when the reactor cavity is dry could also result in a puff release of fission products to the environment if it occurs during containment venting or after containment failure.

5.2.6 Eliminate Water in Reactor Cavity Before VB

Ex-vessel steam explosion at VB has been identified in NUREG-1150 as a significant mechanism for causing early drywell failure and SP bypass for a Mark III containment. Methods to prevent, or reduce, early reactor cavity flooding include the avoidance of an upper pool dump before VB, the prevention of hydrogen combustion in the containment, the operation of the drywell-containment vacuum breakers, and the initiation of early containment venting up to the time of VB.

Figure 4.5 shows the impact of the various mitigative actions on the drywell water inventories for a Mark III short-term SBO sequence. Figure 4.5 shows that a significant amount of SP water will overflow the weir wall to the drywell even with controlled hydrogen burns using the hydrogen ignition system. Figure 4.5 also shows that while the operation of the containment-drywell vacuum breakers can reduce the amount of water overflow, significant amount of water overflow still occurs, and containment venting is the single most effective action to reduce the amount of SP water transferred to the drywell. An upper pool dump has not been considered in the calculations of Figure 4.5 because ac power, which is required to open the dump line valves, is not available for the SBO sequence. Significant increase in water overflow would occur should the upper pool be dumped.

The elimination of water in the reactor cavity is in direct conflict with the strategy of flooding the reactor cavity discussed in the previous section. The decision on whether early reactor cavity flooding should be promoted or suppressed depends on the relative beneficial and adverse effects of these two strategies. There are significant uncertainties on the magnitude of an EVSE and on its effect on containment structural integrity. Uncertainty also exists on the effectiveness of water on CCI progression. In general, it seems that having water in the reactor cavity is more desirable. Even if a severe EVSE does occur and fails the drywell, the containment, if not failed before VB, is unlikely to be challenged by the EVSE loads (see Figure 4.4).

Figure 5.1 shows the effects of reactor cavity flooding on the probabilities of the various APBs discussed in NUREG-1150 (See Figure 4.2 for APB categories.). It shows the change in APB probabilities due to SP water overflow. The results were obtained from APET analyses to quantify the effects of the various containment performance improvements on release profiles [12]. The two cases that are used to construct the data presented in Figure 5.1 are similar in all other aspects except that in one case SP water overflow is allowed (same as in the NUREG-1150 analyses) while in the other it is prohibited. Figure 5.1 shows that early containment failure probability (Categories 1 through 4) is not changed by allowing water overflow, but the probability of early SP bypass (Categories 1 and 2) is increased. This is as expected because an overflow increases the probability of EVSE, resulting in drywell failure and early SP bypass. Figure 5.1 also shows that water overflow causes a decrease in the probability of late containment failure (Category 5), a slight decrease in the probability of containment venting (Category 6), and an increase in the probability of containment survival (Category 7).

The results on risk for these two cases are presented in Figure 5.2. The data presented in Figure 5.2 are their ratios to the base case results presented in NUREG-1150. Improvements to the base case considered in these two cases include an improved HIS with 100% diffusion burn efficiency, an improved post-core-damage reactor depressurization capability, and an increased probability of using the fire water system for low pressure injection. Figure 5.2 shows a reduction of risk in all categories (i.e., mean early fatalities, mean latent fatalities, mean 50-mile dose, mean 1,000-mile dose, and mean offsite costs) by SP water overflow.

The above results seem to indicate that the strategy to flood the reactor cavity is more desirable than the strategy of eliminating water from the reactor cavity. However, in the analyses discussed above SP water overflow is not permitted over the whole accident duration covered by the APET. Risk may be reduced if the reactor cavity can be kept dry before VB but flooded later after the corium is on the floor, if this can be managed. However, later addition of water to the hot corium in a dry cavity may still result in a vigorous fuel-coolant interaction (FCI) and significant loading condition. The progress of CCI will also be more severe in the dry-cavity case than in the case of flooding the cavity before VB. The net effect on the risk is still uncertain at this time. More research work may be needed to provide data for a better management of this issue.

5.2.7 Combustible Gas Control

The amount of combustible gases that can be generated from both in-vessel core degradation and ex-vessel CCI is significant and may create an atmosphere that exceeds deflagration/detonation limits. Combustible gas control is primarily achieved by the use of the hydrogen ignition system (HIS), i.e., to burn the combustible gases at low concentrations to avoid a rapid containment pressurization (a quasi-static load) due to deflagration or a dynamic load due to detonation. If the HIS is not working and containment gas composition reaches unacceptable limits, the challenges of combustion may be mitigated by a containment vent and purge operation, or simply by containment venting. Containment spray can condense steam and de-inert a steam-inerted containment and thus increase the challenge of combustion, but it can also be used with the HIS to mitigate the effect of combustion.

The unavailability of the HIS during SBO sequences, the dominant plant damage states (PDSs) for a Mark III containment, and the potential of failing both the containment and the drywell by combustion make combustible gas related challenges the most risk significant challenges to Grand Gulf. According to existing EOPs, the HIS is manually energized when the RPV level drops to below the TAF level. Existing EOPs also instruct the operator not to turn on the HIS if power to the system is not available, or uncertain. This is to avoid inadvertently providing an ignition source at power recovery.

To reduce the risk associated with the burning of combustible gases, modifications to the existing HIS have been suggested, and their effects on risk analyzed [12]. The possible modifications include backfitting the current ac-powered HIS with an independent power supply or installing a catalytic ignitor system that can operate without power. This improvement will ensure the availability of the HIS during SBO sequences and is most effective in combustible gas control for short-term SBO sequences, the most dominant severe accident sequences for Grand Gulf. The effectiveness of this improvement for long-term SBO sequences is not as significant, primarily because the containment is most likely steam-inerted when the concentration of combustible gases is high.

The cost and effectiveness of installing backup dc power to the ignitors for a Mark III containment has been analyzed [12]. The improved HIS was found to produce a significant decrease in the conditional probability of early containment failure for Grand Gulf. Development work on catalytic ignitors is being carried out by Sandia National Laboratories in the United States and by Siemens in Germany [39].

The operation of the HIS during a long-term severe accident may not prevent a steam-inerted containment from having a high combustible gas concentration. Detrimental combustion events may occur later, after the steam is condensed. Controlled combustion is important for combustible gas control in these cases. One method to control combustible gases is to energize selected HIS ignitors to initiate burns in selected volumes at an atmospheric composition that would not cause unacceptable results. An example is the controlled burning of combustible gases in the drywell [12]. The drywell purge system can be used to control the drywell atmospheric composition such that unacceptable burns would not occur. Another method to control the burning of combustible gases is by a controlled and coordinated operation of both the HIS and the containment spray [40]. The containment spray can condense steam and allow the burning of combustible gases before their concentration reaches dangerous levels.

Controlled combustion requires a good knowledge of the atmospheric composition in the various volumes of the containment. Instrument indications for such information are available for hydrogen and oxygen concentrations in both the containment and the drywell. However, these instrument indications may not be available during a station blackout because of a lack of power supply. Samples of the containment atmosphere can be obtained and analyzed during such an accident to provide data on containment gas concentration. The usefulness of this method depends on the time required to obtain this information and the speed of accident progression.

Containment composition can also be controlled by the use of the containment vent and purge system if the containment pressure is within the range of operation of this system. Containment venting, although it cannot change the composition of the containment atmosphere, can be used to reduce containment initial pressure and the amount of oxygen and combustible gases in the containment and thus the effect of combustion. Containment venting, when the containment is steam-inerted, can also reduce the amount of oxygen in the containment and possibly prevent later combustion. The venting path should be reclosed in time to prevent the possibility of a negative pressure load on the containment, or re-introduction of oxygen into the containment, after steam condensation. Since sufficient amounts of noncondensible gases are generated during both in-vessel core melt and ex-vessel CCI, the removal of oxygen from the containment does not necessarily cause a deficiency of noncondensible gases in the containment. The optimum control of containment venting requires a good knowledge of the containment atmosphere composition and the progress of the processes that affect this composition, e.g., the progress of CCI and the rate of steam condensation.

Detrimental combustion events can also be avoided if an ignition source is not introduced when the containment composition is at a dangerous level, and systems that may cause a rapid steam condensation and thus de-inert the containment are not actuated. The former involves a careful control of energizing the HIS, and the latter involves the control of the containment spray and the upper pool dump. Early hydrogen burn, before the RPV breaches, can be mitigated by reducing the amount of in-vessel hydrogen production, by controlling the time of RPV depressurization (Section 4), and a severe containment hydrogen burn shortly after VB can be avoided by RPV depressurization if the HIS is operating (Section 5.2.2).

The combustion due to steam condensation of a hydrogen-rich but steam-inert containment may not result in damaging containment loading, if the rate of steam condensation is low. Combustion in the containment is initiated as steam concentration drops to below its inerting level if an ignition source is available. The extent of the combustion, and the resultant containment loading, depend on the completeness of the combustion, which may be influenced by the condensation rate and the steam concentration level at combustion initiation. Uncertainty exists in this area. While the combustion completeness model in HECTR and MELCOR depends only on the initial concentration of the combustible gases, the combustion completeness model in MAAP is more complicated (using an analytical model) and depends on the initial steam concentration level. If the extent of combustion is limited by the initial steam fraction and the resultant load is not detrimental, it is then desirable to provide an ignition source such that combustion can occur when steam concentration reaches the combustible limit. The HIS, if available, can be used to provide such an ignition source for a controlled combustion. However, additional research effort is needed to determine the effect of initial steam concentration on combustion completeness and to assure that the resultant loading is not detrimental.

5.2.8 Flooding a Leak Area for Fission Product Scrubbing

Should the containment failure mode be a leak, fission product release can be reduced if the leak location can be identified and flooded. The leakage will then pass through a pool of water and some of the fission products will be retained. Where applicable, this strategy can be used to reduce fission product release to the environment from the containment atmosphere or directly from the RCS.

The secondary containment radiation and temperature monitoring systems can be used to identify the leak area. Analytical results on containment performance, such as those presented in NUREG-1037 can provide information about potential leak areas and the ways to flood these areas.

Even a moderate leak area would result in a significant volumetric flow rate [32], consequently, the operating staff must determine quickly, after the leak is identified, whether the area can be flooded, the means to flood the area, and any potential adverse effects.

5.2.9 Strategies Related to Secondary Containment Fission Product Retention

In the event of primary containment failure fission products are usually discharged to the secondary containment (SC) before they pass into the environment. The fission product retention capability of the secondary containment is not considered in the NUREG-1150 Grand Gulf analyses because the upper part of the SC, the enclosure building, is a very weak structure, and is expected to rupture at containment failure. However, there are cases when the systems in the secondary containment can be used to reduce the amount of fission product release to the environment. The effectiveness of these systems depends on the mode of containment failure. They are expected to be more effective if the containment failure is a leak such that the flow rate to the SC is small.

The SC systems that could be used for further FP release reduction are the Standby Gas Treatment System (SGTS) and the fire spray system. These systems and their ability to retain fission products are discussed below.

Using the SGTS to Reduce Fission Product Release: The BWR EPGs call for the SC Heating, Ventilation, and Air-conditioning (HVAC) system to be isolated and the SGTS to be initiated when the secondary containment HVAC exhaust radiation level exceeds its isolation setpoint. The SGTS is used to remove fission products from the secondary containment atmosphere by HEPA and charcoal filters and to discharge the effluent from an elevated location, e.g., an off-gas stack. The height of the release point (above ground) varies from 32 ft for Grand Gulf to 200 ft for Clinton.

The designed discharge capacity of the SGTS is small when compared with the expected containment leak rates. However it may still be possible to operate the SGTS without damaging the system even when the containment leak rate is much greater than the SGTS capacity. This is because the secondary containment is not designed to withstand a significant pressure load and substantial leakage will develop in the secondary containment as pressure increases. SGTS operation under this condition will still be beneficial because part of the leak flow will pass through the filters of the SGTS and be released at an elevated location. Even if the HEPA filters of the SGTS fail from aerosol plugging, their charcoal bed adsorption efficiency may still be maintained. Even when both HEPA and charcoal filters fail, the operation of the SGTS may still be desirable because of the elevated release point it provides. On the negative side the operation of the SGTS may reduce the residence time of some fission products, and thus their retention, in the secondary containment. This adverse effect becomes important if the filters of the SGTS have failed and the secondary containment pressure is low (i.e., a low containment leak rate).

Secondary Containment Fire Spray for Fission Product Retention: The secondary containment fire spray system can also be used to reduce the release of radioactive materials if the system can be actuated manually. Similar to the containment spray system, the fire spray system can remove airborne fission product aerosols and vapors by the mechanism of impaction, interception, Brownian diffusion, diffusiophoresis, and thermophoresis.

The fire system for Grand Gulf has one electrically driven pump and two diesel driven pumps. Each pump has a capacity of 1,500 gpm at 125 psig. Since the diesel driven pumps do not depend on plant electric power, this strategy could be available during SBO sequences if the ability to manually operate the fire system exists.

The need for fission product removal from the secondary containment atmosphere can be inferred from high offsite radioactivity and high secondary containment area radiation readings. However, the secondary containment instrumentation may not be working properly because the environmental conditions may be harsher than that for which the equipment is qualified.

The fire spray will condense the steam in the secondary containment and increase the possibility of an early hydrogen burn in the secondary containment.

5.2.10 Other Strategies

The other strategies listed in Table 5.2 are strategies that have been included in the BWR EPGs and in some cases involve the designed usage of the systems, e.g., containment and SP cooling. The challenges these strategies can mitigate are shown in Table 5.1. These strategies have been discussed in Section 4.

Table 5.1 Challenges and the Parameters to Identify Challenges

		Parameters	
Challenge		Direct	Indirect
1.	Loss of Resources	Instrument indication for Electric Power, Water and Pneumatic Supply	Functionality of Plant Systems
2.	SP Dynamic Loads		RPV Pressure, Containment Pressure, and SP Temperature and Water Level
3.	Containment Pressure	Containment Pressure	
4.	Drywell Temperature	Drywell Temperature	
5.	Ex-Vessel Steam Explosion		RPV Breach Condition, Reactor Cavity Condition
6.	Rapid Drywell Pressurization Related to HPME		RPV Breach Condition, Reactor Cavity Condition
7.	Burn of combustible Gases	Cont. Hydrogen and Oxygen Concentrations	
8.	Core-concrete Interaction (CCI)		Containment Pressure, Temperature, Gas Concentrations, and FP (Radiation)
9.	FP in Containment Atmosphere	Containment Radiation	
10.	FP From Containment to Outside	Secondary Containment (SC) or Environment Radiation	Containment Radiation
11.	FP From RCS to Outside	SC and Environment Radiation	Containment Radiation
12.	FP From SC to Environment	SC and Environment Radiation	

Table 5.2 Strategies and the Challenges Addressed by the Strategies

		Challenge											
	Strategy	Loss of Resources	SP Dynamic Loads	Containment Pressure	Drywell Temperature	Ex-vessel Steam Explosion	Rapid Drywell Pressurization Related to HPME	Burn of Combustible Gases	Core-concrete Interaction (CCI)	FP in Containment Atmosphere	FP From Containment to Outside	FP From RCS to Outside	FP From SC to Environ-ment
1	Resource Management	+											
2	RPV Depressurization		+				+	(+)		+		+	
3	Containment Venting			+	$(-)^{(2,3)}$	+	(+)	(+)			-		
4	Containment Spray			+				$(\pm)^{(4)}$		+			
5	Reactor Cavity & Containment Flooding				+	-	(+)		+	+			
6	Eliminate Water in Reactor Cavity Before VB					+			-	-			
7	Combustible Gas Control							+					
8	Leak Area Flooding										+		
9	SC FP Retention												+
10	Other Strategies												
	Containment Cooling			+						+			
	SP Cooling		+							+			
	DW Purge				+					(+)			
	SP Water Level Control		+										
	Isolate Leak Area										+	+	

Note: 1. + Mitigating Effect
2. - Potential Adverse Effect
3. ( ) Secondary or Uncertain Effect
4. ± Effect Depending on Conditions

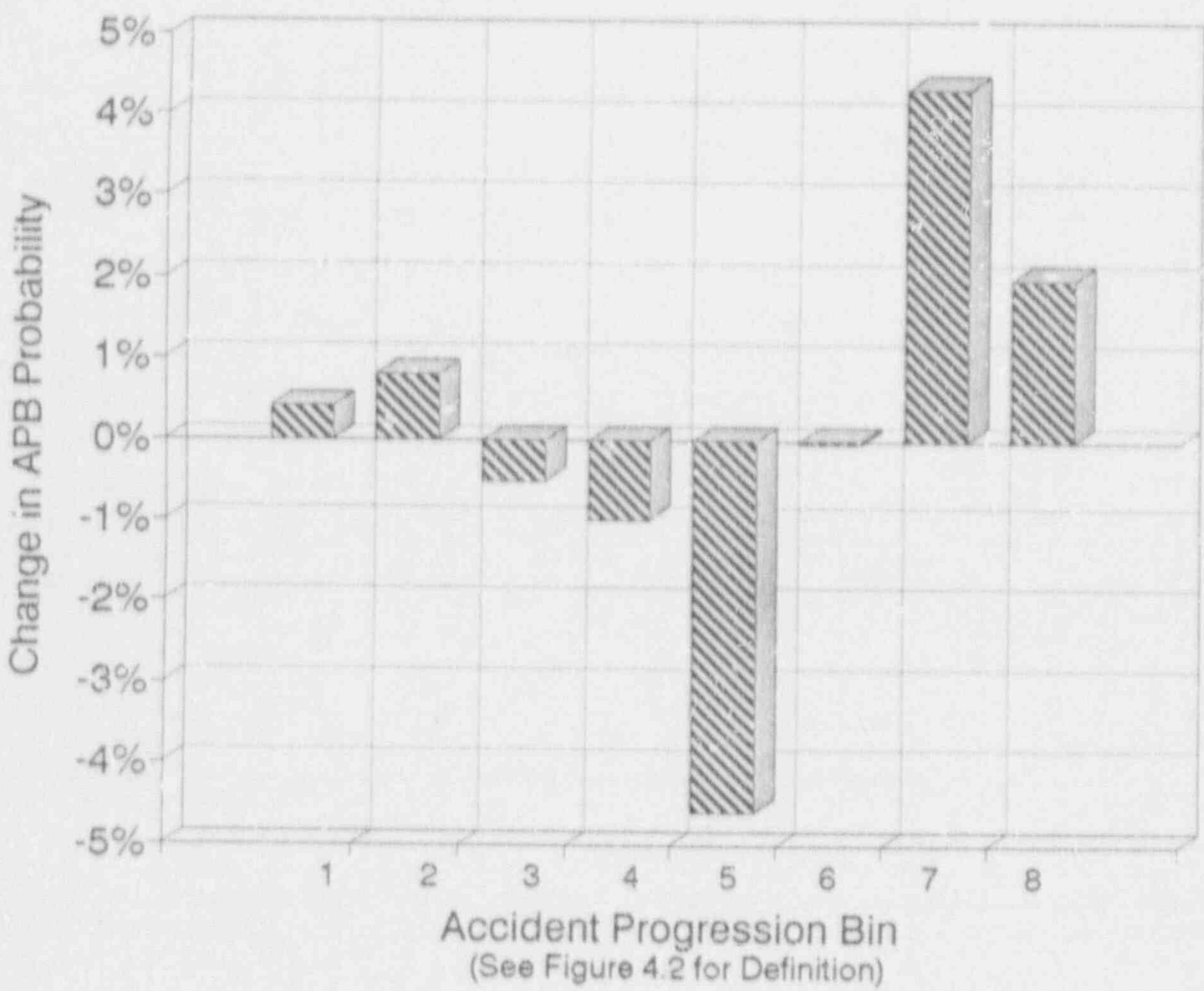

Figure 5.1 Change in APB Probability by Allowing Weir Overflow (Reference 12)

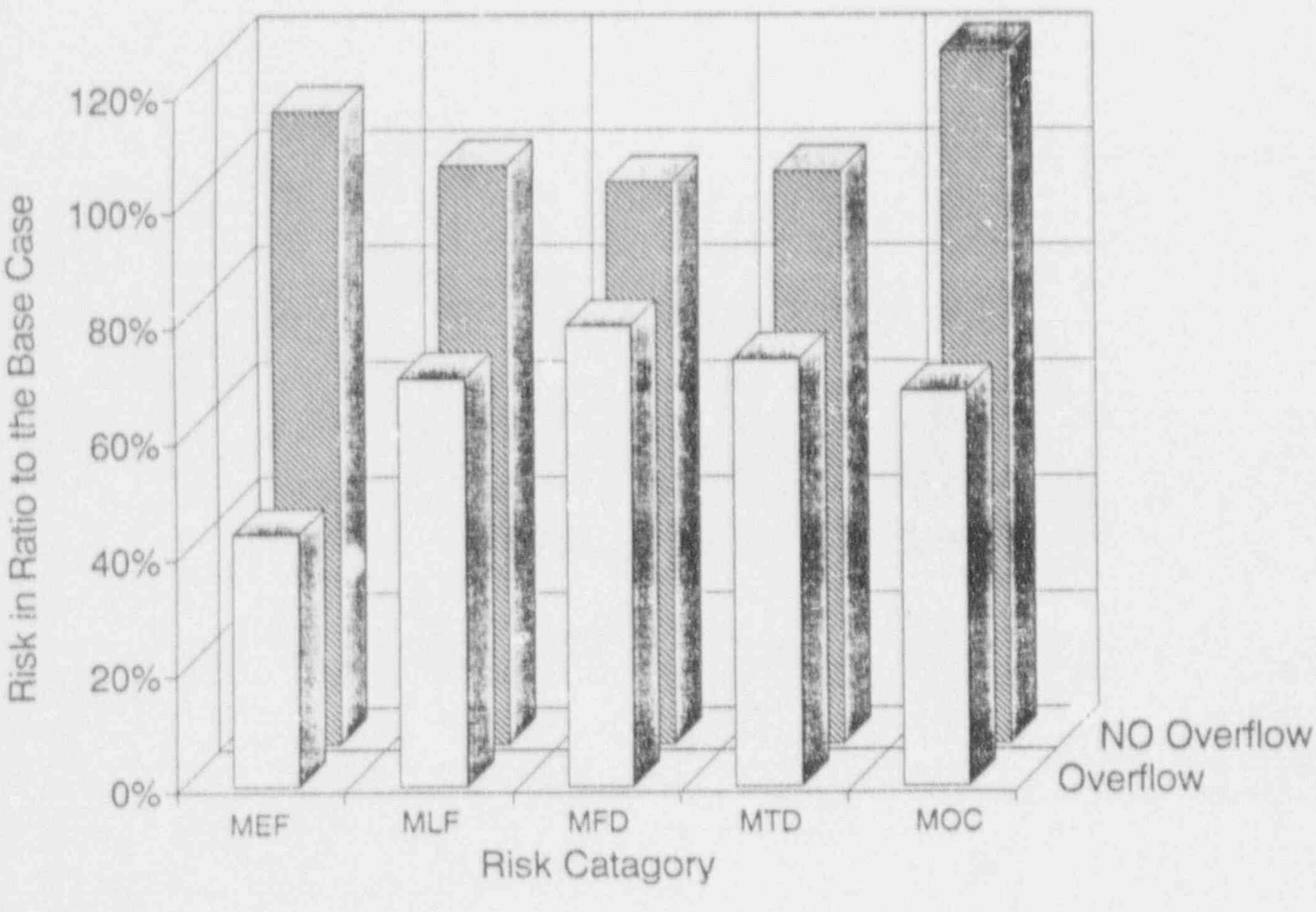

Figure 5.2 Change in Risk With and Without Weir Overflow (Reference 12)

6 Strategy Application to Accident Sequences

In this section the strategies are assessed by application to certain accident sequences. For strategy assessment each sequence is divided into the phases described in Section 2.3. Under each phase the expected challenges are discussed, the strategies which can address these challenges are applied, and the effects of implementing accident management strategies evaluated.

The Grand Gulf Nuclear Station was used as the surrogate plant for this assessment. The Grand Gulf EOPs, instead of the BWR EPGs, were used to determine the operation response as currently expected at the plant.

6.1 Severe Accident Sequence Selection

The selection of sequences used in the strategy assessment process requires engineering judgement and should fulfill several objectives. The sequences selected should among them cover all the identified challenges and thereby allow all the strategies to be considered. At the same time sequences with a high probability of core damage or with high consequences should obviously be considered. Especially the latter need to be included in the assessment of containment and release strategies. Multiple failures of safety systems should also be treated.

The sequence categories selected consisted of station blackout, ATWS, loss of containment heat removal, and isolation failure. These provide a range of accident characteristics which need to be considered: the initial condition of the reactor and the containment at the inception of the accident, the speed of accident progression, and the availability of major safety systems.

Selection of the above sequences should not be construed as implying that the identified strategies are only applicable to the sequences discussed. The strategies will often be beneficial under other conditions as well, although the circumstances surrounding those conditions may need to be accounted for in the strategy implementation.

6.2 Station Blackout Sequences

Station blackout (SBO) sequences are initiated by a loss of off-site power and all three diesel generators, i.e., the diesel generators for both Division 1 and 2 emergency power and the dedicated diesel generator for High Pressure Core Spray (HPCS) system. This leads in a Mark III BWR to the loss of all active engineered safety features except the steam powered Reactor Core Isolation Cooling (RCIC) system. Since the RCIC system requires dc power for control, it fails after the depletion of station batteries. The RCIC turbine may also trip because of high turbine exhaust pressure (i.e., containment pressure), which occurs due to the loss of containment heat removal. The loss of all core injection would result in core damage, vessel breach, and eventual containment failure, if recovery and mitigative actions are not successful.

The SBO sequence where all core injection is lost at the beginning of the accident and core damage occurs early, at about one hour after accident initiation, is termed a fast SBO sequence (or short-term SBO sequence). A fast SBO sequence may be caused by the loss of all dc power, or simply a failure of the RCIC system, in addition to the loss of all ac power. Fast SBO is the dominant plant damage state for Grand Gulf. It contributes approximately 94% of the total core damage frequency. A slow SBO sequence (or long-term SBO sequence) is a sequence where core injection (e.g., by RCIC) is available initially and core damage begins at about 12 hours after accident initiation (typical time for battery depletion). Slow SBO contributes approximately 3% of the total core damage frequency. The combined contribution of all SBO sequences is about 97% (Table 4.1).

6.2.1 Characteristics of SBO Sequences

SBO sequences are characterized by the loss of most of plant instruments and equipment. The availability of containment instrument indication after the loss of ac power, or the loss of both ac and dc power, would be minimal, and is plant dependent. The plant also loses the systems for containment heat removal (CHR) (e.g., containment and SP cooling), and the ability to deliver water to the containment (e.g., containment spray). The most important accident management activities after a station blackout therefore should be (1) to recover ac power, (2) to extend dc power, and (3) to identify and utilize alternate systems and resources. More detailed discussions of these issues have been provided in Section 5.2.1 for resource management. Other strategies discussed in Section 5 may also help to prevent or mitigate the effect of the challenges that may occur during the progression of the accident. Detailed discussions are presented below.

6.2.2 Containment Response to SBO Sequences

In a fast SBO sequence, all core injection is lost at accident initiation. The water in the RPV is boiled off by the reactor core decay heat and the steam generated in the RPV is discharged to the suppression pool through the SRV lines. The steam is condensed in the suppression pool, causing the suppression pool temperature to rise. The containment, in turn, is pressurized by water evaporation from the suppression pool. Because of the large amount of water in the suppression pool and the large volume of the containment, the suppression pool will remain subcooled and the containment pressure and temperature will remain low before core damage begins.

Because of a longer duration of core injection and boil-off, the containment pressure before core degradation may be very high in a slow SBO sequence. Containment design pressure (15 psig) could be reached for a 12-hour core injection, and the estimated containment failure pressure (55 psig) may be reached during an 18-hour core injection. The suppression pool will be saturated in both of the above cases.

Core damage begins at about two to four hours after the loss of core injection. The generation and release of steam and hydrogen in the RPV will cause further containment pressurization. There is uncertainty in the amount of hydrogen generated in-vessel before vessel breach. The MAAP code predicts small in-vessel hydrogen production, primarily because of the modeling of refreezing of the molten cladding in the lower part of the channel and the resulting blockage of the channel for steam passage and Zircaloy-steam reaction [14]. On the other hand, STCP and MELCOR predict much higher in-vessel hydrogen production [9, 11, 13]. The amount of in-vessel hydrogen production predicted by STCP and MELCOR will cause the containment hydrogen concentration to exceed the hydrogen flammability limit, or even the detonation limit.

The vessel fails at about two hours after core melt begins. The drywell will experience a significant quasi-static pressure load if vessel breach occurs at high reactor pressure. The pressure load associated with the high pressure melt ejection (HPME) may fail the drywell and cause a suppression pool bypass. The pressure increase in the containment after the high-pressure vessel breach is not significant, however, even if the suppression pool is bypassed [7]. The drywell may also experience a significant load at vessel breach even if the vessel fails at low pressure; the interaction between the molten core debris and the water in the reactor cavity may cause a quasi-static pressure load on the drywell and a dynamic load on the reactor pedestal.

The core-concrete interaction (CCI) occurring after vessel breach will generate additional steam and noncondensible gases, resulting in further containment pressure rise. Without operator actions, the containment pressure will continue to rise until containment failure pressure is reached.

Containment temperature will not be significant (in general below 250°F) if the suppression pool is not bypassed and a hydrogen burn does not occur. The hot gases generated from both in-vessel core melt and ex-vessel CCI will pass through the suppression pool and are cooled before entering the containment. Drywell temperature will in general be higher. The drywell atmosphere is heated primarily by the RPV surface before vessel breach. If there is a stuck-open vacuum breaker on the SRV tail pipe, hot gases from the RPV can enter the drywell directly and cause the drywell temperature to rise even more rapidly before vessel breach. After vessel breach, the drywell atmosphere is heated by the hot gases released from CCI. The drywell temperature may exceed 1,000°F if the CCI is allowed to proceed in a dry reactor cavity.

It should be noted that significant uncertainties exist in the understanding of some severe accident phenomena, and in the ability to predict containment responses accurately. However, the results of the STCP and MELCOR calculations can be used to identify the important features of containment response in an SBO sequence. They will be used as a basis for the discussion of the challenges and strategies presented below.

6.2.3 Challenges and Strategies During SBO Sequences

Table 6.1 shows the challenges occurring in a fast SBO as well as the strategies and SAM actions required to mitigate these challenges. The times shown in Table 6.1 are based on the results obtained from STCP and MELCOR calculations [9, 11, 13]. Corresponding information for the slow SBO sequence is shown in Table 6.2 (based on six-hour battery life). These two tables are used to guide the discussions that follow.

6.2.3.1 Challenges and Strategies in the Very Early and Early Phases

With the loss of both offsite and onsite ac power, an Alert emergency class would be declared at the beginning of a slow SBO sequence, and with the additional loss of dc power or RCIC, a General Emergency would probably be declared at the beginning of a fast SBO sequence (NUREG-0654). This declaration of emergency classes will trigger the activation of the TSC and entry into the emergency procedures. The TSC is expected to be operational within about 30 minutes and will take control of plant operations and provide technical support to reactor operations.

Tables 6.1 and 6.2 show the plant status indications and the corresponding EPG actions. There are no significant containment challenges during the very early phases of a fast SBO sequence. Steam generated in the RPV from decay heat is discharged through the SRVs to the SP and causes a temperature increase in the SP, which, in turn, causes a slow pressure and temperature increase in the containment atmosphere. As noted in Table 6.1 containment actions called for by the EPGs are not likely to be carried out because of the lack of instrument readings or unavailability of required equipment.

The containment may be challenged by containment pressurization during the very early phase of a slow SBO sequence if the core injection is extended for a long time (beyond six hours), but without a corresponding recovery of containment heat removal capability. (The STCP calculation, which is used as the basis for Table 6.2, assumes a six hour battery life.) This may happen if the battery life is extended by load shedding, or if an alternate water source, independent of plant ac and dc power (e.g., diesel-driven fire water), is used for core injection. The challenges will then be similar to those of a sequence involving loss of the containment heat removal capability, which is discussed later in this section.

As shown in Tables 6.1 and 6.2, core melt starts at about two hours after accident initiation for the fast SBO case and about four hours after battery depletion for the slow SBO case. The difference between these two cases is primarily due to the decrease in decay heat with time. After the onset of core melt, containment pressurization is primarily caused by the production and release of hydrogen. The containment venting pressure (PCPL) may be reached for a slow SBO during this phase. However, containment venting may not be carried out because of the lack of ac power.

A significant amount of hydrogen is produced rapidly in the RPV once the core reaches the runaway oxidation temperature for the metal-water reaction. The amount of hydrogen produced in-vessel depends on the timing of RPV depressurization and the availability of water for metal-water reaction. It is in general sufficient to cause the containment hydrogen concentration to reach combustible, or even detonatable, limits before vessel breach. The most important strategy during this time phase is therefore the control of combustible gases. As discussed in the previous sections, the most effective means to control combustible gases is the use of the hydrogen ignition system (HIS). Since the HIS depends on ac power, it is not available during SBO. An alternate and independent power source for the HIS ignitors, or catalytic ignitors, is needed for controlled burning of combustible gas during SBO sequences.

Strategy Application

As mentioned above, once the oxidation temperature for the metal-water reaction (an exothermic reaction) is reached, the in-vessel hydrogen production rate will increase substantially, and as a result, a significant amount of hydrogen is generated and released in a short time. Table 6.1 shows that the containment hydrogen concentration reaches 5%, a level at which hydrogen can be burned in a benign manner, about 20 minutes after core melt starts. Containment hydrogen concentration reaches over 15%, a level where a hydrogen burn is likely to cause detrimental results, in another 20 minutes. The HIS should be started before containment hydrogen concentration reaches 5%, but prohibited after it goes beyond an acceptable limit, e.g., 12%. There is only a narrow time window to energize the HIS once core melt has begun.

The containment atmosphere may be steam-inert in a slow SBO sequence. The steam concentration in the containment depends on containment pressure before core degradation, which in turn, depends on the time duration of core injection. If the containment pressure before VB is 15 psig (the containment design pressure), the containment steam concentration is likely to be between 35% to 50% (depending on containment temperature). The containment will be inert to detonation (35%) but not to deflagration (55%). Hydrogen deflagration is therefore possible. The containment is likely to be inert to both deflagration and detonation (greater than 55% steam concentration) when the containment pressure before core melt is above the containment venting pressure. This condition is likely to occur if core injection has been available for over 12 hours after accident initiation.

Containment steam concentration can be increased by containment venting. The steam released to the containment atmosphere during venting replaces the air rich containment atmosphere vented to the outside and thus increases the steam concentration in the containment. Early venting, before core degradation begins, not only increases containment steam concentration, but also removes oxygen from the containment and thus reduces containment oxygen concentration. Later hydrogen combustion may be avoided (due to oxygen deficiency) if a sufficient amount of oxygen can be removed from the containment.

A large amount of steam is required to reduce the oxygen concentration from its initial value of over 20% to below its combustion limit of %. A containment steam pressure of about 45 psig (assuming normal temperature) is required to dilute the containment oxygen concentration to below 5%. It requires a long period of core injection operation to achieve this. Containment venting, although not effective to reduce containment oxygen concentration, can be used to reduce the amount of oxygen (in kg-moles) in the containment. containment venting at a pressure of 45 psig will reduce the amount of containment oxygen to a value that will remain below 5% at atmospheric pressure. Subsequent containment pressurization by additional steam will reduce the oxygen concentration to below 5%.

The removal of noncondensible gases (i.e., air) from the containment by containment venting may cause a negative containment pressure load (if the vent path is reclosed) or a re-introduction of air (and thus oxygen) into the containment (if the vent path is open) when the containment cools down. The release of in-vessel hydrogen during core degradation can slow down, or even prevent, the containment pressure reduction by supplying additional noncondensible gas to the containment. However, the amount of hydrogen generated during core degradation (in kg-moles) is at most half the amount of the noncondensible gases originally in the containment and may not be sufficient to make up all the noncondensible gases lost from venting.

In addition to the above effects, containment venting also reduces the containment base pressure and thus reduces the challenge to containment integrity even if severe hydrogen combustion does occur. A detailed knowledge of containment atmosphere composition and accident progression is required to control containment atmosphere composition and thus prevent detrimental hydrogen combustion by containment venting. More detailed analyses and improved knowledge on in-vessel hydrogen production are also needed to realize the potential benefits of containment venting on combustible gas control.

Because of the lack of any ignition source, hydrogen combustion may not occur during a fast SBO sequence even if the containment atmosphere reaches combustible limits. It is therefore important not to provide any ignition source if the containment atmosphere composition reaches a dangerous combustion level. This level could be reached in a fast SBO sequence shortly after core melt starts. Possible ignition sources include the HIS, which may be energized by a recovery of ac power if it was turned on before power recovery, or by subsequent operator action.

If the containment is steam-inert and the containment hydrogen and oxygen concentrations are above the limits which may cause significant containment loads, it is important to maintain the inert state and thus prevent hydrogen combustion from occurring. Containment spray, if recovered and actuated at this time, may condense enough steam to render the containment atmosphere flammable. The containment spray is very effective in removing steam from the containment atmosphere. MELCOR and HECTR calculations show that the steam concentration drops by 10 % (from 55% to 45%) after about ten minutes of containment spray operation [9]. An upper pool dump may also condense containment steam and induce a similar result as containment spray. To dump the upper containment pool requires a permissive signal, which is generated by a low RPV water level or a high drywell pressure LOCA signal. This signal will be available early in an SBO sequence as shown in Tables 6.1 and 6.2. The upper containment pool may be dumped automatically 30 minutes (by a delay timer) after the permissive signal if power for the SPMU system valves is available. Operator action may be needed to prevent an upper pool dump if the containment condition is not favorable.

Although the containment atmosphere may be combustible during core melt, the drywell atmosphere will not be combustible for this time duration because in-vessel hydrogen is discharged through the SRVs into the suppression pool, i.e., outside the drywell. Calculations with MELCOR and HECTR indicate that the drywell atmosphere can at most be marginally flammable before vessel breach [9]. However, hydrogen can be released to the drywell directly if there is a stuck-open SRV vacuum breaker. A stuck-open vacuum breaker not only increases drywell hydrogen concentration before VB, but also increases the amount of fission products in the containment atmosphere, because some of the in-vessel fission products released to the drywell will pass through the drywell-wetwell leak paths to the containment atmosphere, bypassing the scrubbing of the suppression pool. The operator can control the operation of the SRVs (by reducing the number of repeated operations of a particular SRV) to reduce the probability of a stuck-open vacuum breaker. The operator can also avoid the use of the SRV with a stuck-open vacuum breaker.

Even though the drywell atmosphere is not combustible during core melt, the use of the drywell purge system can transfer the containment atmosphere to the drywell and thus change the drywell atmosphere composition. This can be used to control the drywell atmosphere for a controlled hydrogen combustion in the drywell if hydrogen combustion in the containment is not safe after power recovery. The drywell purge permissive signal is generated at the same time as the SPMU permissive signal and can be started after a 30 seconds delay. The potential and details of using the drywell purge system for a controlled hydrogen burn in the drywell to consume hydrogen and oxygen in the containment for combustible gas control require further analyses.

Vessel breach occurs at about 3.5 hours after accident initiation for a fast SBO sequence and about 6 hours after the loss of core injection for a slow SBO sequence. The drywell integrity is challenged at vessel breach by a quasi-static pressure load if the RPV is breached at high pressure (HPME), and by a dynamic load if an energetic fuel-coolant interaction (EVSE) occurs. As shown in Tables 6.1 and 6.2 RPV depressurization has been called for before VB by the plant EOPs under several containment conditions. RPV depressurization would also have been called for by the reactor control guidelines of the EOPs. RPV depressurization before a significant amount of hydrogen is produced is also required to remove any concern about loads on the suppression pool boundaries by SRV actuation and pool swell.

The RPV can remain at high pressure before VB due to either the loss of control power or operator errors. Loss of dc power is one of the important initiators for fast SBO plant damage states in NUREG-1150. The enhanced reactor depressurization system as recommended by the CPI program includes a dedicated source of dc power to the SRV solenoids for improved operation of the SRVs during severe accidents. RPV depressurization before VB will eliminate the challenge to drywell integrity associated with HPME. It also reduces the probability of detrimental hydrogen combustion in the containment, because a high pressure vessel breach will discharge a large amount of hydrogen to the containment in a very short time and provide ignition sources to the containment.

RPV depressurization allows the reactor hydrogen to be discharged to the containment in a more gradual manner and be better controlled by the HIS, if operable. It also prevents the discharge of hot reactor gases and core debris at high pressure, which may punch through the suppression pool and thus provide ignition sources to the containment.

An ex-vessel steam explosion (EVSE) generates a dynamic load on the reactor pedestal and pressure loads on the reactor pedestal and the drywell. The probability of EVSE can be reduced by eliminating water from the reactor cavity before vessel breach. The amount of water in the reactor cavity before VB can be reduced, or eliminated, by the operation of drywell-wetwell vacuum breakers, early containment venting, or avoidance of hydrogen burns in the containment (Table 4.5). An early upper pool dump should also be avoided to reduce the potential of water overflow to the drywell and the reactor cavity. As already discussed in Section 5, this strategy requires actions that are in contradiction to other strategies. More detailed discussion of this strategy can be found in Section 5.

6.2.3.2 Challenges and Strategies in the Late Phase

Containment pressure and temperature continue to rise after the accident enters the late phase. As shown in Tables 6.1 and 6.2, containment conditions would exceed some EOP limits (e.g., 15 psig containment pressure and 330°F drywell temperature), but some of the actions required are not relevant any more (e.g., RPV depressurization). Containment cooling and containment spray, if available, can be used to remove energy from the containment atmosphere and thus reduce pressure and temperature loads. Containment venting can be used to remove both mass and energy from the containment atmosphere and thus reduce the pressure. The latter action is the only means to reduce the pressure increase due to the noncondensible gases resulting from CCI. As shown in Table 6.1 and 6.2, the containment venting pressure (PCPL, 17.25 psig for Grand Gulf) is reached 8 hours after accident initiation for the fast SBO sequence and about 11 hours after accident initiation for the slow SBO sequence. However, containment failure pressure (55 psig for Grand Gulf) is not reached until about 40 hours after accident initiation (STCP predicts a failure time of about 20 hours and MELCOR and MAAP predict a failure time of more than 40 hours).

The most significant challenges to containment integrity during this phase are hydrogen combustion in the containment, if the hydrogen in the containment has not been burned before, and the progression of CCI. CCI will release noncondensible gases to the containment atmosphere and cause a containment pressure increase which cannot be controlled by containment cooling or containment spray. CCI may also cause a significant temperature load on the drywell and erosion of the reactor pedestal and basemat. The strategy to control the progression of CCI is therefore important.

The actions for combustible gas control during this phase are the same as those discussed above for the early phase. The additional noncondensible gases produced by CCI will reduce the probability of negative containment pressure or re-introduction of oxygen to the containment if containment venting has been used for combustible gas control.

Since the sources of combustible gases are in the drywell now, there is a time window when the drywell atmosphere may reach combustible, or even detonatable limits, for a fast SBO sequence. MELCOR calculations show the drywell atmosphere composition is in a range to support deflagration and detonation from 3.5 to 7 hours after accident initiation [13]. In reality, the combustible gases may burn as they leave the surface of the core debris and may not accumulate in the drywell to a detonatable level. Accumulation of combustible gases in the drywell may continue after the oxygen in the drywell is consumed by the combustion.

Flooding the reactor cavity before vessel breach and continuously adding water to the corium increases the probability of slowing down the progress of CCI, and eventually terminating it. The strategies of reactor cavity and containment flooding are important for this phase. One action that can be taken to increase the potential for reactor cavity flooding is to start the upper pool dump early in the accident, before core melt begins. The upper pool dump can be effected at about one hour after the initiation of a fast SBO sequence if the valves of the discharge line can be opened using an alternate power source. The decision to initiate an upper pool dump in a slow SBO sequence is more complicated because of its effect on steam condensation and hydrogen combustion. Additional discussion on reactor cavity and containment flooding is provided in Section 5.

6.2.3.3 Challenges and Strategies in the Release Phase

The challenges during this phase, the times when they occur and the mitigating strategies are shown in Table 6.1 for a fast SBO sequence and in Table 6.2 for a slow SBO sequence. General discussions of the challenges and strategies for fission product release have been presented in detail in Section 4.2.4. Individual strategies have been discussed in Section 5, and an assessment of the effectiveness of plant systems to mitigate FP release, e.g., containment spray and pool scrubbing, has been given in Section 3 in terms of their decontamination factors for FP removal.

Containment failure may occur in the very early phase, before core melt begins, for a slow SBO sequence if core injection has been operating for a long time (e.g., over 18 hours) while containment heat removal capability is not available. The amount of fission products released to the environment after a very early containment failure would not be significant if core melt is prevented. The primary severe accident management effort at this time is therefore to prevent core melt, or arrest further core degradation if core melt has already begun (by in-vessel management strategies). For cases where core melt cannot be arrested and vessel breach is imminent, the primary accident management effort would then be shifted to preserve drywell integrity. The drywell would most likely remain intact after containment failure because the slow containment pressurization process that fails the containment is not likely to challenge the drywell. An intact drywell will assure a pool scrubbing of the fission products before they are released to the containment.

After a very early containment failure, the only mechanisms that may challenge drywell integrity as the accident progresses are the loads associated with HPME and EVSE in the short term, immediately after vessel breach, and the loads associated with CCI and reactor pedestal erosion in the long term. Hydrogen combustion is not likely to occur because the containment atmosphere is most likely to be steam-inert. The strategies discussed above for mitigating the effects of the above challenges can be used to reduce the probability of drywell failure. One strategy that merits special attention is the strategy to flood the containment. Containment flooding can be used to cool the RPV as an in-vessel strategy. It also provides a deep pool for cooling the core debris (and thus delays or prevents CCI), and for scrubbing fission products released from CCI if it occurs. The extended time of core injection that causes the containment overpressure failure may have added a significant amount of water to the suppression pool if water source external to the containment has been used for core injection. The external water sources include the condensate storage tank with a capacity of 300,000 gallons and the two fire water tanks with a capacity of 300,000 gallons each. These water sources can be used to flood the containment. Additional discussions on containment flooding can be found in Sections 4 and 5.

If the containment has not been failed in the very early phase, if may still be failed during core degradation, i.e., after core melt but before VB. The cause of containment failure during this time frame is primarily hydrogen deflagration or detonation. The transient pressure load from hydrogen combustion may fail both the containment and the drywell at the same time. Containment hydrogen combustion is more likely to occur in a fast SBO sequence than in a slow SBO sequence because the containment atmosphere is likely to be steam-inert for the latter case. The containment atmosphere may reach a flammable limit about two hours after accident initiation (Table 6.1). However, as discussed previously, there is uncertainty regarding the amount of in-vessel hydrogen production. The MAAP code predicts a small amount of hydrogen production during core degradation and as a consequence hydrogen combustion may not be possible during this time frame. If a significant hydrogen combustion event does occur during core degradation and fails both the containment and the drywell, the most important accident management effort is to preserve the integrity of the RPV. It is also important to prevent a release from the RPV to the drywell (e.g., through a stuck-open SRV vacuum breaker) because the suppression pool is bypassed for this release. If the accident proceeds to vessel failure, it is then important to depressurize the vessel before VB to assure that the fission products released from the vessel would pass through the suppression pool and be scrubbed. Containment spray, if available, can also be used to scrub fission products from the containment atmosphere. Its use is particularly valuable if there is a suppression pool bypass, such as in the cases where there is a stuck-open SRV vacuum breaker or a vessel failure. The effectiveness of containment spray for fission product scrubbing has been discussed in Section 3.2.3.

If the hydrogen combustion event fails only the containment, but not the drywell, and the accident proceeds to vessel breach, the important CRM effort is then to preserve the integrity of the drywell. The challenges and strategies that have been discussed above for the very early time phase will apply here.

If the integrity of both the containment and the drywell is preserved before vessel breach, the energetic events (HPME and EVSE) occurring at VB may challenge the drywell, as well as provide additional hydrogen and ignition sources to the containment. The containment and drywell failure probability at VB is significant for SBO sequences (Figure 4.4). Because of the speed of these energetic events, actions should be taken before VB to be effective. The strategies discussed in Section 5 can be used to prevent these energetic events from occurring, or mitigate their effects if their occurrence cannot be prevented. If the energetic events at VB fail only the drywell, the fission products will be retained in the containment and the release to the environment will be insignificant. The most important CRM effort will then be maintaining containment integrity. Containment spray can be used for fission product scrubbing if its operation will not cause any hydrogen combustion concern. On the other hand, if only the containment is failed, the important CRM effort would then be maintaining the integrity of the drywell. Strategies discussed above for maintaining drywell integrity after containment failure will apply.

If the containment survives the challenges in the early phases of the accident, it may still fail in the late phase by slow containment pressurization, due to energy and gases (mostly noncondensible gases) released from CCI, or by hydrogen combustion. The time of containment failure is an important factor determining the severity of fission product release (Section 3.1.1.2). As shown in Tables 6.1 and 6.2, containment failure by slow containment pressurization will occur more than 40 hours after accident initiation. However, the containment may fail earlier if hydrogen combustion occurs in the containment.

After a late containment failure, one important CRM effort is to maintain the integrity of the drywell. The drywell may be challenged in the late phase by the temperature load from the hot gases released from CCI. In addition to being a source of the drywell temperature load, CCI is also the source of ex-vessel fission product release, containment pressure load, and combustible gases. The most important strategy at this time is therefore the control and mitigation of CCI. This may be achieved by flooding the reactor cavity. Containment flooding, an extension of reactor cavity flooding, is a very effective strategy for controlling accident progression and fission product release at this time (Sections 4.2.3 and 5.2.5). Drywell temperature load, if not alleviated by CCI control, can be mitigated by the operation of the drywell purge system, if it is available and there is a sufficient pressure head to prevent leakage from the drywell to the containment through this system. Containment spray, if its operation does not cause any concerns about gas combustion, can be used for additional fission product scrubbing.

Additional strategies for mitigating the release of fission products to the environment include the strategies to control fission product revolatization and late release of iodine from water pools, the strategy to flood the leak areas, and the strategies for fission product retention in the secondary containment. These strategies have been discussed in Sections 4.2.4, 5.2.8, and 5.2.9.

6.3 ATWS Sequences

The ATWS sequences discussed in this section are those initiated by an MSIV closure at full power. Reactor power, after a successful automatic recirculation pump trip and RPV water level control, would still exceed the containment heat removal (CHR) capability of the RHR system. The discharge of the RPV steam to the suppression pool would lead to a rapid suppression pool heat up and containment pressure rise. The containment would fail if recovery or mitigative actions are not successful. Since the ECCS pumps for Mark III containments have been designed to pump saturated water, they are not lost at containment failure (an assumption used in previous ATWS calculations [10, 14]). Core injection is postulated to be lost due to random hardware faults and operator error (e.g., failure to depressurize the RPV). The loss of all core injection leads to core melt and vessel breach.

In a fast ATWS sequence, core injection is lost early such that core damage occurs in the short term. Core melt and vessel breach occur prior to containment failure. In a slow ATWS sequence, core injection is lost late so that core damage, core melt, and vessel breach occur after containment failure. In NUREG-1150 analyses, core damage for both ATWS sequences is driven by the operator's failure to depressurize the RPV, and, as a result, the vessel is at high pressure during core degradation. The low pressure injection is recoverable if the RPV is depressurized.

6.3.1 Characteristics of ATWS Sequences

ATWS sequences are characterized by the significant amount of thermal power generated in the core and released to the containment. The primary objectives of operator actions are to (1) reduce core power by in-vessel strategies, and (2) increase the energy removal capability from the RPV and/or from the containment. Energy can be removed from the RPV by restoring the main condenser as the heat sink (an in-vessel strategy), or from the containment by venting (an ex-vessel strategy). The major concerns for containment venting are whether there is sufficient venting capacity to remove the input thermal power and whether there is sufficient time to complete the venting actions. These and other issues regarding venting have been discussed in Section 3.

Since ac power is available during an ATWS sequence, most of the important plant systems and instruments discussed in Section 3 would be available during the accident. Some of the plant systems and instruments may be lost during the accident due to harsh environmental (e.g., containment temperature and pressure) or loading (e.g., loads associated with HPME or hydrogen combustion) conditions.

6.3.2 Slow ATWS Sequences

6.3.2.1 Containment Responses of a Slow ATWS sequence

Because of the high reactor power, the suppression pool temperature rises rapidly in the early phase of an ATWS sequence. The suppression pool will be saturated in about 20 minutes after accident initiation. (Actual time depends on the power level controlled by in-vessel strategies.) Containment pressure will rise due to suppression pool evaporation, and containment failure pressure will be reached in about one to two hours after accident initiation. Table 6.3 shows the timing of key events during the accident and the challenges and strategies (or SAM actions) required for the sequence.

Table 6.3 is based on a STCP calculation of the ATWS sequence where core injection is assumed to fail as a result of containment failure [10]; and core melt starts at about a half hour after the loss of core injection. If core injection continues after containment failure, the time to core melt can be extended accordingly.

Vessel breach occurs at about 2 to 3 hours after the loss of core injection, with actual time depending on the time of injection loss and the corresponding decay heat level. The drywell will be challenged at VB by the energetic events occurring at VB (i.e., HPME and EVSE). Drywell temperature may rise to over 1,000°F a few hours after vessel breach due to CCI.

The control of reactor power by in-vessel strategies, e.g., RPV level and pressure control, will affect the energy input rate to the containment and consequently the pressure and temperature responses in the containment. This will influence the vent area required for containment pressure control, the time of occurrence of major phenomenological events, and the time available for ex-vessel responses.

6.3.2.2 Challenges and Strategies of Slow ATWS Sequences

Table 6.3 shows the challenges and the strategies and the SAM activities required for a slow SBO sequence. It is used to guide the discussion that follows.

6.3.2.2.1 Challenges and Strategies in the Very Early and Early Phases

With the failure of the reactor protection system to initiate and complete a scram, an Alert would be declared at the beginning of an ATWS sequence. A Site Area Emergency may also be declared. The TSC and the EOF would be activated early in the accident and the TSC is expected to be operational in about 30 minutes after accident initiation.

As shown in Table 6.3, the EPG control variables would exceed their limits within about an hour after accident initiation because of the large energy input rate to the containment. The operator actions required to respond to these conditions include RPV depressurization, drywell cooling, drywell spray, and containment venting. RPV depressurization is assumed not to be carried out due to operator error and the low pressure core injection systems are therefore not available for core cooling. Containment cooling and containment spray (if operated with RHR heat exchangers) would remove some energy from the containment, but their combined capacity is most likely to be below the energy input rate. However, with the addition of containment venting there maybe the capacity to handle the energy input rate and prevent containment failure for many ATWS cases.

The venting area required to keep the containment pressure below the failure pressure depends on the net energy input rate to the containment atmosphere and the pressure loss along the flow path. Figure 6.1 shows the effective venting area required to keep the containment pressure at a constant value for various energy input rates. (The curves in Figure 6.1 are based on isentropic choked flow of dry steam taken as an ideal gas.) For a particular net energy input rate, the containment pressure will increase and stabilize at a higher pressure if the vent area is too small, and containment failure may occur if this steady state pressure is above the containment failure pressure. Figure 6.1 shows that an effective vent area of about 2 ft^2 is required to maintain containment pressure below 55 psig for a net energy input corresponding to 10% rated thermal power of the reactor. The effective vent area could be much smaller than the nominal area of the vent path. Factors accounting for the actual valve opening area, the pressure loss along the flow path, and the effect of actual composition and real gas properties should be considered in the determination of the effective vent area.

Containment venting is the most important strategy to prevent a very early containment failure. Containment venting pressure (PCPL) for a Mark III containment is 17.25 psig. This pressure may be reached in approximately 30 minutes to one hour after accident initiation. Since ac power is available, there seems to be sufficient time to open the vent path (described in Section 3.2.4). The important issue is then whether the vent area is sufficient to maintain the containment pressure at PCPL. Since PCPL is relatively low for a Mark III containment, the vent area required to maintain the containment pressure at (or below) PCPL will be relatively large. The 20-inch vent exhaust line (Grand Gulf CVS) itself may not be sufficient to handle a containment steaming rate corresponding to a reactor power typical for an ATWS sequence (e.g., greater than 10%). Containment pressure will then increase and stabilize at a higher pressure, or containment failure pressure will be reached. If containment pressure continues to rise after the opening of the 20-inch exhaust line, another 20-inch line, the supply line of the containment ventilation system, can also be opened. The two 20-inch lines would appear to be able to maintain the containment pressure below the containment failure pressure if the reactor power can be reasonably controlled (e.g., less than 15% rated power). However, the real capacity of the flow lines needs to be assessed. The real capacity (or effective venting area) of the venting lines is influenced by the actual valve opening area and the friction loss in the flow path.

There are other plant systems that can be used to reduce the containment pressure rise rate during an ATWS sequence. The containment cooling system, in either the suppression pool cooling mode or the containment spray mode, can be used to remove some of the energy dumped to the suppression pool. The SPMU system can also be used to increase the water volume of the suppression pool and thus the energy absorption capability of the pool. An upper pool dump can add approximately 40% normal suppression pool water volume in a very short time (less than 10 minutes for Grand Gulf). This will increase the suppression pool energy absorption capacity by about 40%, and delay the time to reach the containment failure pressure by about 40%. The data presented in table 6.3 indicates that there is sufficient time to carry out an upper pool dump before containment failure. According to the data in Table 6.3, the time to containment failure will be extended by more than 30 minutes. This will provide additional time for corrective actions.

Shortly after containment venting, the containment atmosphere would be practically void of noncondensible gases and become full of steam. The use of a high capacity, cold containment spray in this steam environment, after the vent path has been closed, could cause a rapid containment pressure decrease and an unacceptable negative pressure load, and should therefore be avoided. The vent path can be reopened to supply air to the containment after the reactor power is under control and the containment is depressurized.

After the loss of core injection, reactor water will be boiled off and core melt will begin. The reactor power will be reduced to decay heat level after the core is uncovered. The progression of the accident and the challenges and strategies will be similar to those for a long term SBO sequence. However, since ac power is available in the ATWS sequence, the plant systems required for accident mitigation will be readily available.

6.3.2.2.2 Challenges and Strategies in the Late and Release Phases

Without successful containment venting, the containment could fail before significant core degradation takes place. With the containment failed, the primary objectives of CRM in the later phases are to control the progress of CCI and to reduce the release of the fission products to the environment. Since the core power is reduced to its decay power level after core melt, the challenges and strategies of the release phase of the accident will be similar to those of the SBO cases discussed in Section 6.2.3.3.

6.3.3 Fast ATWS Sequences

6.3.3.1 Containment Responses of a Fast ATWS Sequence

In a fast ATWS sequence, core injection is lost early. The reactor power is high and the containment pressure rises rapidly before the loss of core injection. After the loss of injection, the reactor power is reduced to its decay value as the RPV loses its water level, and the energy input rate to the containment is greatly reduced. The containment pressure when core melt begins depends on the time when core injection is terminated. The hydrogen released to the containment during core melt is burned in the containment by the operation of the HIS if the containment is not steam inert. The energetic events at high pressure vessel blowdown (HPME/DCH and EVSE) will result in a significant load on the drywell. The hydrogen produced from these energetic events and the hydrogen released from the vessel at VB will be transported to the containment and a severe hydrogen burn event may occur in the containment following VB. The load from the hydrogen burn may challenge both the containment and the drywell.

If the containment has not failed, containment pressurization will continue during CCI. Drywell temperature may rise to over 1,000°F a few hours after failure due to CCI.

6.3.3.2 Challenges and Strategies of Fast ATWS Sequences

The challenges to containment integrity for a fast ATWS sequence are similar to those for a fast SBO sequence except that because of the availability of ac power, the plant systems, particularly the HIS, are available for the ATWS sequence. The probability of containment failure before VB (primarily by hydrogen burn in the containment) is greatly reduced in the fast ATWS sequence because of the operation of the HIS.

The most important challenge to early containment failure is hydrogen combustion in the containment following VB. The hydrogen released from the vessel at VB and the hydrogen produced in the drywell in HPME/DCH and/or EVSE (by metal/water interaction) will be transported to the containment and cause a hydrogen combustion in the containment (some of the hydrogen may be burned in the drywell, but most of the hydrogen will be transported to the containment). Although there are uncertainties, the amount of in-vessel hydrogen production and the amount of hydrogen production by HPME/DCH and EVSE can be significant and the subsequent hydrogen combustion can be severe. A severe hydrogen combustion in the containment may challenge both the containment and the drywell. Of the ten most probable APBs presented in NUREG/CR-4551, the containment fails immediately after VB in eight of the ten APBs, and the drywell fails at the same time in two of these eight APBs.

The most important strategies to prevent early containment failure is to remove the mechanisms that cause the challenges. RPV depressurization, if achieved before VF, allows the low pressure injection system to deliver water to the vessel and may arrest core degradation and prevent vessel breach. Even if core degradation is not arrested, RPV depressurization will eliminate HPME and remove the associated challenges. EVSE can be prevented by eliminating water from the reactor cavity. The benefits and adverse effects of eliminating water from the reactor cavity have been discussed in Sections 5.2.5 and 5.2.6. Containment spray can also be used to mitigate the effect of a containment hydrogen burn.

If the containment and the drywell survive the challenges occurring immediately after VB, the challenges to containment and drywell integrity in the late phase of the accident arise primarily from CCI. Challenges and strategies discussed in Section 6.2.3.2 for the late phase of the SBO sequences will apply in the present case. The challenges and strategies for fission product release control would also be similar to those discussed in Section 6.2.3.3 for SBO sequences. The availability of electric power during an ATWS sequence would improve the chances for implementation of some strategies.

6.4 Loss of Containment Heat Removal Sequences

Accidents involving the loss of long term containment heat removal (CHR) (e.g., TPI sequence in Reference 3 and T_{23}QW sequence in Reference 13) are similar to the slow ATWS sequences discussed in Section 6.3.2 in terms of the sequence of major events (e.g., vessel breach and containment failure) and the failure mode of the RPV and the primary containment. There is a net energy increase in the containment for both types of accidents, and the containment will fail by overpressure if corrective actions are not taken. Since the net energy input to the containment in a loss of CHR sequence is at the decay heat level, which is much lower than the energy involved in an ATWS sequence, the containment pressure increase is much slower and the time available for operator action is consequently much longer.

The operator actions required to mitigate the various challenges of a loss of CHR sequence are similar to those of a slow ATWS sequence (Table 6.3), but with a much longer time scale. The most important operator action, containment venting, is not required until more than 10 hours after accident inception. The capacity of the containment venting area is also not a concern because of the low power level in this sequence. The probability of successful containment venting is therefore very high. This reduces the significance of the loss of CHR sequences, and as a result, this sequence does not contribute significantly to the total core damage frequency of Grand Gulf in NUREG-1150 [4].

If containment venting is not successful, or if all reactor core makeup is lost, the accident will progress to core degradation and subsequent vessel breach. The progression and CRM measures will be similar to that of a slow ATWS sequence discussed in Section 6.3.2.

6.5 V Sequences

V sequences involve the breach of the reactor coolant system (RCS) pressure boundary at an interface with a low pressure system. The rupture of the low pressure system outside the primary containment and the unavailability of the core coolant makeup systems (which may be a consequence of the rupture) lead to a core melt and the release of fission products directly to the secondary containment. The V sequence is not included in the plant damage states (PDS) considered in NUREG-1150 due to the low core damage frequency (CDF) associated with this type of sequence. However, since the release bypasses the primary containment and the suppression pool, it is a high risk sequence and will be discussed here.

There is no appreciable pressure and temperature increases in the containment before vessel breach because the break is outside the primary containment. The primary system loses its water inventory through the break area, and core degradation and vessel breach will follow after the depletion of all core water.

The blowdown of the high temperature radioactive steam from the RPV directly to the outside, bypassing the primary containment, will result in a high temperature radioactive atmosphere in the area near the break and an entry condition for the secondary containment control procedure (based on BWR EPGs). Following the instructions in the procedures, the operator will try to isolate the systems that are discharging into the high temperature area. If this fails to control the secondary containment conditions, the operator is then instructed by the procedures to shutdown the reactor, enter RPV control guidelines, and perform emergency RPV depressurization.

The secondary containment area radiation level would also exceed the operating limit as the accident proceeds. The operator actions for area radiation level control are similar to those for area temperature control discussed above. It should be pointed out that high secondary containment area temperatures may also be caused by a fire in that particular area. However, with the instrumentation indication available for the RPV and for the area radiation monitoring system, it should not be difficult to distinguish one from the other.

Isolation of the system that leaks to the outside of the primary containment could eliminate the source of the accident and terminate its progression, or it could change the sequence to one that loses core injection but without containment bypass, similar to an SBO sequence discussed in Section 6.2. If the break area cannot be isolated, RPV depressurization would reduce the driving force for the break flow and thus the amount of release to the outside. (The pressure in the RPV before vessel breach may remain high, even with the leak.) During RPV depressurization, some of the gases and the fission products generated in the RPV are discharged to the containment through the SRV lines and the SP; the total flow and fission products leaked to the outside of the primary containment are thus reduced. Since significant amounts of fission products are generated in the RPV after the start of core melt (about 30 minutes after accident initiation), it is desirable to initiate emergency depressurization as soon as possible, and to maintain the SRVs open throughout the accident. The EPGs are adequate to address release control as discussed above.

The release could also be reduced by flooding the pipe that leads to the leak area or by keeping the leak area submerged under water. This is very plant specific, and the identification of potential leak areas and the preplanning for possible means to flood these potential leak areas are important for the success of this strategy.

After vessel breach, the pressure in the RPV would be in equilibrium with that in the containment. The pressurization of the containment from CCI would drive the containment atmosphere through the RPV and the leak area to the outside of the containment. By passing through the RCS, the fission product release may be enhanced because of RPV revolatilization. The radioactivity from this mode of release can be reduced by reducing the containment pressure. This can be achieved by containment venting. The flow to the secondary containment from containment venting will have the benefit of fission product scrubbing by the SP, and the total release will therefore be reduced. In fact, if the leak area cannot be isolated, it may be desirable to open all available containment vent paths as early as possible to have the maximum amount of the release pass through the SP. Since the PCPL is not reached and the radioactivity released would exceed the operating limit, containment venting is not likely to be carried out based on the EPGs.

After vessel breach, the containment atmosphere will leak to the outside of the containment through the RCS. Containment integrity is therefore lost after vessel breach. The primary objective of accident management is then to control the progress of CCI and the release of fission products. The challenges and strategies for this case are similar to those for the SBO cases discussed in Section 6.3 (Tables 6.1 and 6.2).

Table 6.1 Challenges and Strategies for a Fast SBO Sequence

Time Hr:Min	Plant Status	Challenge (Comment)	Strategy or SAM Actions
0:00	**Loss of AC and All Core Injection**	(Plant Damage State)	Alert & Site Emergency Declared General Emergency Declared Activate TSC & EOF
		Very Early Phase of the Accident	
			Very Early Recovery Actions
	SP/T > 95 F	(EPG Entry Condition)	Monitor and Control SP/T, SP/L, PC/P, DW/T and CN/T (Note 1) *SP Cooling (Note 2)* *Containment Cooling*
	DW/T > 135 F	(EPG Entry Condition)	
0:15	*SP/T > 110 F*	Potential Unacceptable SP Boundary Load	Enter EPG RPV Control G/L (Note 3)
0:30	*SP/T > 120 F*	(TSC Operational) Potential Unacceptable SP Boundary Load	RPV Depressurization
	RPV Level 1 LOCA Setpoint	(Permissive Signal for SPMU with 30 minute delay)	
		Early Phase of the Accident	
			Early Recovery Actions Emergency RPV Depressurization
1:40	**Core Melt Starts** *SP/T > HCTL*	Potential Containment Overpressure	
	PC/P Increase & VB Imminent	SP Boundary Load Potential Challenge at VB	*Early Venting* *Reactor Cavity Flooding* *Eliminate Water*
2:00	*Hydrogen Concentration > 5%*	(Hydrogen Ignition Limit)	*Combustible Gas Control*
2:10	*Hydrogen Concentration > 12%*	Detrimental Combustion Potential	*Combustible Gas Control*
3:30	**Vessel Breach**	Load Associated with HPME (DCH, M&E Addition, Hydrogen Combustion, and SP Load)	RPV Depressurization or Above Actions Before VB to be Effective
		Ex-vessel Steam Explosion	*Eliminate Water*

Table 6.1 Challenges and Strategies for a Fast SBO Sequence (Continued)

Time Hr:Min	Plant Status	Challenge (Comment)	Strategy or SAM Actions
		Late Phase of the Accident	
			(Note 4)
3:30	DW/T > 330 F	Containment Temperature Load	Late Recovery Actions Containment Spray Emergency RPV Depressurization
6:00	PC/P > 9 psig	Potential Containment Overpressure and SP Boundary Load	Containment Spray Emergency RPV Depressurization
7:30	PC/P > 15 psi	Containment Pressure Load	RPV Depressurization
	CCI	Pressure, Temperature Loads & Noncondensible Gas Generation	Reactor Cavity Flooding Containment Flooding
8:00	PC/P > PCPL	Containment Pressure Load	Containment Spray Containment Venting
40:00	**Containment Failure**		
		Release Phase of the Accident	
1:40 - 3:30	In-vessel Release and Release at VB	FP in Containment Atmosphere	RPV Depressurization
3:30	Ex-Vessel Release (CCI) Revolatilization Late Iodine Release All Above	(1.5 - 4 Hours After VB)	Reactor Cavity Flooding Water to RPV and Cooling SP Cooling Containment Spray and Flooding
40:00	Containment Failure or Venting	FP Release from Containment to Outside	Flooding Leak Area Containment Flooding
	FP & Pressure in Secondary Containment	FP Release from Secondary Containment to Environment	SGTS Fire Spray

Note:
1. SP/T - Suppression Pool (SP) Temperature, SP/L - SP Level, PC/P - Drywell or Containment Pressure, DW/T - drywell Temperature, CN/T - Containment (Wetwell) Temperature.
2. Letters in italic indicate that the information or system may not be available because of lack of support, e.g., electric power.
3. The RPV control guideline should have been entered earlier.
4. The availability of instruments and equipment during a SBO is very uncertain, particularly after battery depletion. Unless special arrangements have been made, they are generally not available. However, recovery of E/P will make some of them available.

Table 6.2 Challenges and Strategies for a Slow SBO Sequence

Time Hr:Min	Plant Status	Challenge (Comment)	Strategy or SAM Actions
0:00	**Loss of Offsite & Onsite Power**	(Plant Damage State)	Declare Alert Emergency Activate TSC
		<u>Very Early Phase of the Accident</u>	
			Very Early Recovery Actions
	SP/T > 95 F	(EPG Entry Condition)	Monitor and Control SP/T, SP/L, PC/P, DW/T and CN/T (Note 1) *SP Cooling (Note 2)*
	DW/T > 135 F	(EPG Entry Condition)	*DW Cooling*
0:15			Declare Site Area Emergency Activate EOF
0:30		(TSC Operational)	
1:00	SP/T > 110 F	Potential Unacceptable SP Boundary Load	Enter EPG RPV Control G/L (Note 3)
1:30	SP/T > 120 F	Same as Above	RPV Depressurization
3:00	SP/T > HCTL (150 F at System Pressure)	Same as Above	Emergency RPV Depressurization
3:30	PC/P > 1.23	(EPG Entry Condition) (Permissive Signal for SPMU with 30 minute delay)	*DW Cooling and Containment Spray*
6:00	Battery Depletion	(Partial Recovery Will Change the Sequence to a TW Sequence)	General Emergency Declared Now or Earlier
7:00	PC/P > 9 psig	Potential Containment Overpressure SP Boundary Load (Containment Steam Inert for Detonation)	*Containment Spray*

Table 6.2 Challenges and Strategies for a Slow SBO Sequence (Continued)

Time Hr:Min	Plant Status	Challenge (Comment)	Strategy or SAM Actions
		<u>**Early Phase of the Accident**</u>	
10:00	**Core Melt Starts**		Early Recovery Actions
	RPV Repressurization	SP Boundary Load	*Alternate E/P & P/S (Note 4)*
10:20	*Hydrogen Concentration > 5%*	(Hydrogen Ignition Limit)	*Combustible Gas Control*
10:30	*Hydrogen Concentration > 12%*	Potential for Detrimental Combustion	*Combustible Gas Control*
	PC/P > 15 psig	Containment Pressure Load	RPV Depressurization
11:00	PC/P > PCPL	Containment Pressure Load	Containment Spray Containment Venting
	PC/P Increase & VB Imminent	Potential Challenge at VB	*Containment Spray, Early Venting* *Reactor Cavity Flooding* *Eliminate Water*
12:00	**Vessel Breach**		
		Load Associated with HPME (DCH, M&E Addition, Hydrogen Burn, & SP Load)	*RPV Depressurization or Above Actions Before VT to be Effective*
		Ex-Vessel Steam Explosion	*Eliminate Water*
		<u>**Late Phase of the Accident**</u>	
			(Note 5)
12:00			
	D/T > 330 F	Containment Temperature Load	Containment Spray RPV Depressurization
	CCI	Pressure, Temperature Loads & Noncondensible Gas Generation	Containment Spray Containment Venting
22:00	**Containment Failure**		
		<u>**Release Phase of the Accident**</u>	
10:00 - 12:00	In-vessel Release and Release at VB	FP in Containment Atmosphere	RPV Depressurization

Table 6.2 Challenges and Strategies for a Slow SBO Sequence (Continued)

Time Hr:Min	Plant Status	Challenge (Containment)	Strategy or SAM Actions
12:00 -	Ex-Vessel Release (CCI) Revolatilization Late Iodine Release All Above	(1.5 - 4 Hours After VB)	Reactor Cavity Flooding Water to RPV and Cooling SP Cooling Containment Spray and Flooding
22:00 -	Containment Failure or Venting	FP Release from Containment to Outside	Flooding Leak Area Containment Flooding
	FP & Pressure in Secondary Containment (SC)	FP Release from SC to Environment	SGTS Fire Spray

Note:

1. SP/T - Suppression Pool (SP) Temperature, SP/L - SP Level, PC/P - Drywell or Containment Pressure, DW/T - Drywell Temperature, CN/T - Containment (Wetwell) Temperature.
2. Letters in italic indicate that the information or system may not be available because of lack of support, e.g., electric power.
3. The RPV control guideline should have been entered earlier.
4. Maintaining RPV depressurization using alternate electric power (E/P) and pneumatic supply (P/S) as recommended by CPI.
5. The availability of instruments and equipment after battery depletion is very uncertain. Unless special arrangements have been made, they are generally not available. However, recovery of E/P will make some of them available.

Table 6.3 Challenges and Strategies for a Slow ATWS Sequence

Time Hr:Min	Plant Status	Challenge (Comment)	Strategy or SAM Actions
0:00	**MSIV Closure & Fail to Scram**	(Plant Damage State)	Alert Emergency Declared Site Emergency Probable Activate TSC & EOF
	Very Early Phase of the Accident		
			Very Early Recovery Actions
	SP/T > 95 F	(EPG Entry Condition)	Monitor and Control SP/T, SP/L, PC/P, DW/T and CN/T (Note 1) SP Cooling
	SP/T > 110 F	Potential Unacceptable SP Boundary Load	Enter EPG RPV Control G/L (Note 2)
	SP/T > 120 F	Same As Above	RPV Depressurization
	DW/T > 135 F	(EPG Entry Condition)	DW Cooling
0:15	PC/P > 1.23 psig	(EPG Entry Condition) (Permissive Signal for SPMU with 30 minute delay)	DW Cooling & Containment Spray
	SP/T > HCTL	SP Boundary Load	Emergency RPV Depressurization
		(TSC Operational)	
0:30	PC/P > 9 psig	SP Boundary Load	Containment Spray Emergency RPV Depressurization
	PC/P > 15 psig	Containment Pressure Load	RPV Depressurization
	DW/T > 330 F	Containment Temperature Load	Containment Spray RPV Depressurization
1:00	PC/P > PCPL	Containment Pressure Load	Containment Spray Containment Venting
1:30	**Containment Failure**	Release of Containment Atmosphere	(See Release Phase)

Table 6.3 Challenges and Strategies for a Slow ATWS Sequence (Continued)

Time Hr:Min	Plant Status	Challenge (Comment)	Strategy or SAM Actions
		Early Phase of the Accident	
			Early Recovery Actions
2:00	Core Melt Starts		
	Hydrogen Production	Potential Detrimental Hydrogen Combustion	Combustible Gas Control
	High Pressure VB Imminent	Potential Challenge at VB	Reactor Cavity Flooding, Containment Spray
4:00	Vessel Breach	Load Associated with HPME (DCH, M&E Addition, Hydrogen Burn & SP Load)	RPV Depressurization, Reactor Cavity Flooding & Containment Spray
		Ex-Vessel Steam Explosion	Eliminate Water
		Late Phase of the Accident	
4:00	CCI	Drywell Temperature Loads and Noncondensible Gas and Fission Product Generation	Late Recovery Actions Reactor Cavity Flooding Containment Flooding
		Release Phase of the Accident	
1:30 - 2:00	Containment Failure	Release of Containment Atmosphere	Resource Management
2:00 - 4:00	In-Vessel Release and Release at VB	FP in Containment Atmosphere	RPV Depressurization
4:00 -	Ex-Vessel Release (CCI) Revolatilization Late Iodine Release All Above		Reactor Cavity Flooding Water to RPV & DW Cooling SP Cooling Containment Spray and Containment Flooding
2:00	Containment Failure or Venting	FP Release from Containment to Outside	Flooding Leak Area Containment Flooding
	FP & Pressure in Secondary Containment (SC)	FP Release from SC to Environment	SGTS Fire Spray

Note: 1. SP/T - Suppression Pool (SP) Temperature, SP/L - SP Level, PC/P - Drywell or Containment Pressure, DW/T - Drywell Temperature, CN/T - Containment (Wetwell) Temperature.
2. The RPV control guideline should have been started earlier.

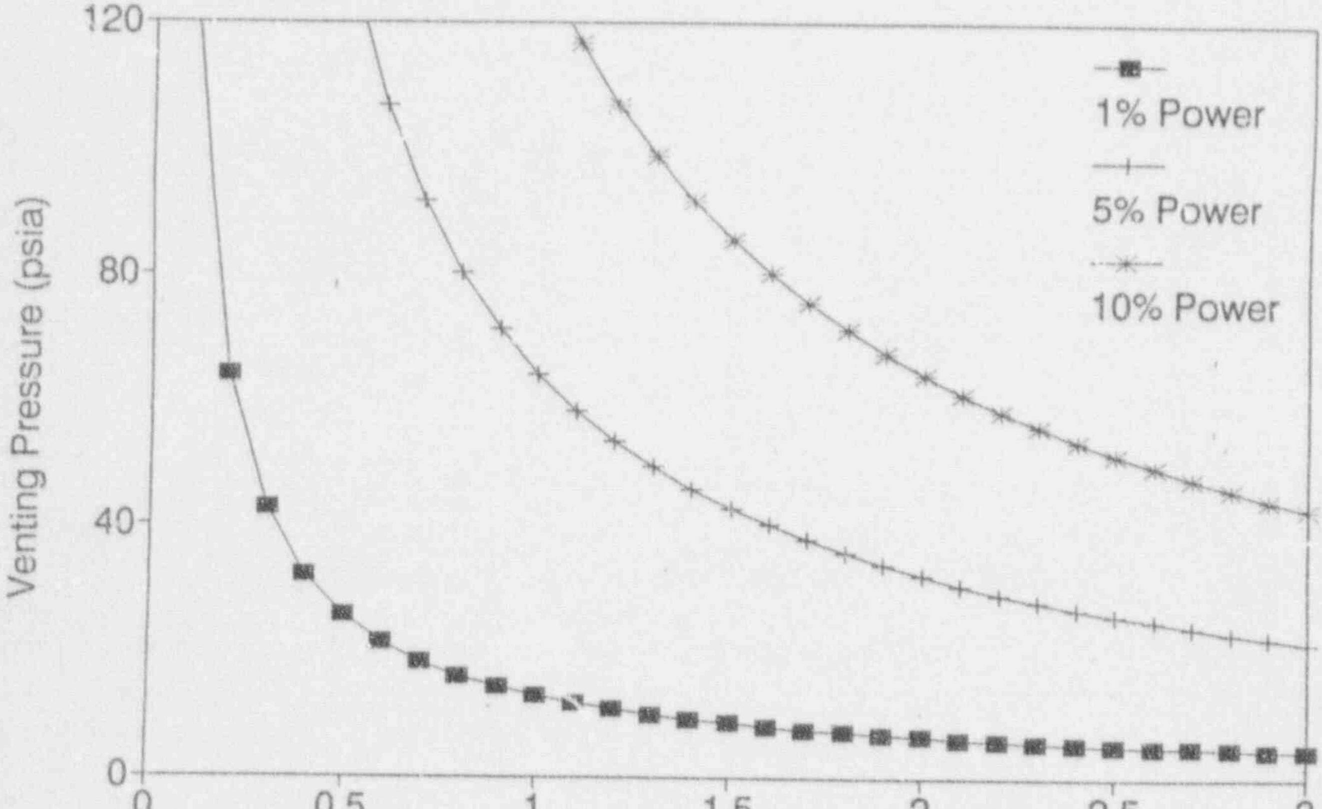

Figure 6.1 Venting Area and Venting Pressure for Various Reactor Power Levels

7 Summary and Conclusions

Information on severe accidents, available from research efforts supported by the NRC under its Severe Accident Research Plan as well as results from the industry sponsored Industry Degraded Core Rulemaking Program, has been reviewed to identify the challenges a Mark III containment 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 from this investigation have been described in this report and are summarized below.

7.1 Existing Accident Management Capabilities

Existing accident management capabilities are based on the NRC requirements described in NUREG-0737 regarding emergency response, and NUREG-0654 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.

The Technical Support Center (TSC): In the accident sequences 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.

Existing Emergency Procedure Guidelines (EPGs): The existing EOPs for a Mark III containment, which are based on the EPGs prepared by the General Electric Company, are symptom based procedures. The plant operations personnel can follow these procedures well into a severe accident by observing the values of some selected plant variables and taking actions accordingly. However, some of the assumptions on which the EPGs 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 EPGs regarding initiating and restricting conditions for accident response actions may be desirable to extend their applicability to accident phases beyond core damage.

The existing EPGs 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.

7.2 Interface Between Existing EOPs and CRM Strategies

As stated in the Introduction of this report, 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 existing EPGs and often rely on the activation of systems designed to cope with design basis accidents. The CRM strategies differ from the existing EPGs 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.

Many actions called for in the EPGs remain valid and useful in the severe accident regime as well. RPV depressurization, for instance, is requested by the BWR EPGs under a number of emergency conditions. If the accident progresses to the severe accident mode before depressurization is implemented, RPV depressurization would still be a beneficial severe accident strategy under almost all circumstances. Another action called for in the EPGs that would generally be beneficial is containment heat removal via suppression pool cooling. However, in this case the question of prioritization, already considered in the EPGs, is also vital in the severe accident phase. If the RHR system is needed for vessel injection or containment spray operation (which provides another mode of containment heat removal), its use for SP cooling may have to be postponed.

Containment venting is another action referred to in the EPGs that has importance as a CRM strategy under severe accident conditions. In the severe accident regime, it may be difficult to proceduralize the initiation of venting. In other words, venting would likely undergo considerable assessment by the TSC at the time of the accident, before it is implemented under some of the severe accident conditions discussed in this report.

Another category of EPG actions are those that will always be beneficial before core damage occurs, but which must be approached with caution under severe accident conditions. Examples for a Mark III containment are strategies involving activation of the hydrogen ignition system (HIS) or the containment spray system. Following the EPG guidance for initiation of these systems may not always be appropriate once a severe accident situation has developed. As pointed out in other sections of this report, operation of the HIS upon recovery from a station blackout requires caution, as does spray operation in a steam inerted atmosphere.

Finally, there are a number of CRM strategies which have no direct counterpart in the EPGs, or, if similar actions are called for in the EPGs, they have a very different basis. Containment flooding is such a strategy, for instance. Flooding is mentioned as a Contingency action in the EPGs as a last resort for vessel, i.e core, cooling. However, the CRM strategy of flooding the containment in anticipation of vessel breach or after vessel failure, has the purpose of achieving CCI mitigation and fission product scrubbing. Another example is the use of the fire sprays in the auxiliary building to mitigate fission product release, resulting from a containment bypass or a failed containment. This is an action not mentioned in the EPGs, but is a strategy that could be used under severe accident conditions.

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.

The containment and release event trees (CRETs) discussed in this report provide a framework for accident progress projection and CRM decision making. The use within the CRETs of current plant status and offsite information together with updated and more accurate estimates of the probability for recovery, etc. can provide a more reliable prediction of the effects of various CRM strategies on accident progression and offsite consequences. Such an approach can be the basis for optimum (in a probabilistic sense) containment and release management (More detailed discussion on this issue can be found in the Mark I report [32].).

7.3 CRM Strategies

The CRM strategies were identified via a detailed examination of the important accident phases using the CRETs as a guide for examination. The identified strategies have been discussed in detail in terms of their applications and potential adverse effects. The strategies have also been assessed by applying them to some accident sequences to determine their feasibility and practicality.

The results of the strategy identification effort are summarized in the safety objective tree (SOT) in figure 4.6. The important strategies identified by this investigation are presented in Table 5.2. Although some of the strategies (e.g., containment venting) have already been included in the existing EPGs, their applications in this report have been expanded to mitigate the challenges that may occur in a severe accident. The strategies in Table 5.2 not included in the existing EPGs are primarily those associated with the energetic events that may occur at VB (HPME/DCH and EVSE) and fission product release reduction.

The most important CRM strategy for a Mark III containment is the control of combustible gases. Combustible gases can accumulate in a Mark III containment to a dangerous level if the HIS is not operating, as occurs in an SBO event, or if the containment is steam-inert, as can occur in a long-term sequence. An improved HIS, with either a dedicated power supply or additional catalytic ignitors, will eliminate the challenge of hydrogen burns in short-term sequences, and will be a very significant improvement because short-term SBO is the most dominant PDS in Mark III containments. However, an improved HIS cannot prevent hydrogen accumulation during a long-term steam inerted sequence. A significant hydrogen burn may occur later in the accident after the steam is condensed, either by the operation of containment cooling systems or by natural cooling through containment structures. Actions are therefore required for combustible gas control in long-term sequences. The feasibility of using HIS, containment venting, or containment venting and purging, has been discussed in this report, but more detailed analyses are required for a successful implementation of this strategy.

According to NUREG-1150 analyses, the HIS is not effective in preventing containment and/or drywell failure due to the burning of hydrogen released from the energetic event following vessel breach (i.e., HPME/DCH or EVSE). The fast metal-water reaction occurring during these events may generate a significant amount of hydrogen in a short time. Subsequent hydrogen burns in the containment may be severe enough to challenge the integrity of the containment and the drywell. Although HPME can be avoided by depressurizing the vessel before VB, the strategy to avoid EVSE, namely, eliminating water from the reactor cavity before VB, involves more uncertainty and questions.

An important issue for Mark III CRM strategies is the conflicting actions required to mitigate the effect of different challenges. The action that is required to mitigate the effect of one particular challenge may have a significant adverse effect on other challenges. Most notable is the action to eliminate water from the reactor cavity. While this action may prevent an EVSE, it also increases the probability and severity of CCI and the subsequent ex-vessel fission product release. There is great uncertainty on the probability and significance of EVSE. Analyses based on current understanding of the phenomena seems to indicate that having water in the reactor cavity would have an overall beneficial effect on the risk profile (Section 5.2.6). Further investigation may be needed to make an optimum CRM decision.

Containment spray, an important plant safety feature, has many beneficial effects for CRM, but may also cause significant adverse effects under certain conditions. Containment spray may cause an unacceptable negative pressure load on the containment, or cause a detrimental hydrogen combustion in the containment. The management of containment spray is therefore important.

Additional examples of conflicting effects of CRM actions include the fission product release from containment venting, and the increase in Alpha mode failure from RPV depressurization. Although an Alpha mode failure for low RPV pressure has been assigned a 1% mean probability in NUREG-1150 analyses, it is believed that the real probability of an Alpha mode failure for a Mark III plant is most likely to be smaller.

The decision to carry out a strategy during a severe accident, particularly those strategies with significant adverse effects depends on balancing the potential adverse consequences of strategy implementation against the consequences that could result if the strategy is not implemented. An integrated approach, such as the use of CRETs discussed above, can be utilized to provide data for decision making. The probability of making the right decision will be increased if the uncertainties can be reduced. Important areas of uncertainties include our current understanding of containment performance, and the ability to predict accident progression accurately. For some strategies further investigation at this time may not be warranted until phenomena are better understood. Even when these uncertainties are resolved as best as possible, there will be a need to consider an optimum choice of strategies. The optimum choice will depend on the impact of a strategy on a particular challenge, as well as on other challenges that may occur concurrently or at later times. As severe accident phenomena are understood better, it should become increasingly worthwhile to investigate and re-evaluate such optimum choice of strategies.

7.4 An Integrated Approach for CRM

CRETs have been used in this report as a guide in the examination of accident sequences for challenge and strategy identification. The same tree structure, with appropriate probability distributions assigned to the individual elements of the tree, can be used to quantity the effectiveness of individual strategies. Another application of the CRETs for accident management is in the prediction of accident progression during an actual accident. When combined with a simple consequence prediction code and with the meteorological conditions and offsite activities already available, this could provide an integrated approach for accident progression and consequence prediction.

8 References

1. "Assessment of Severe Accident Prevention and Mitigation Features," Brookhaven National Laboratory, NUREG/CR-4920, Vol. III, July 1988.

2. "Severe Accident Insight Report," Brookhaven National Laboratory, NUREG/CR-5132, April 1988.

3. "Assessment of Candidate Accident Management Strategies," Brookhaven National Laboratory, NUREG/CR-5474, March 1990.

4. "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, October 1990.

5. "Integration Plan for Closure of Severe Accident Issues," USNRC, SECY-88-147, May 1988.

6. "A Systematic Process for Developing and Assessing Accident Management Plans," Idaho National Engineering Laboratory, Inc., NUREG/CR-5543, March 1990.

7. "Evaluation of Severe Accident Risks: Grand Gulf, Unit 1," Sandia National Laboratories, NUREG/CR-4551, Volume 6, Rev. 1, July 1989.

8. "Analysis of Core Damage Frequency: Grand Gulf, Unit 1 Internal Events," Sandia National Laboratories, NUREG/CR-4550, Volume 6, Rev. 1, August 1989.

9. "MELCOR Analyses for Accident Progression Issues," NUREG/CR-5331, January 1991.

10. "Radionuclide Release Under Specific LWR Accident Conditions," Battelle Columbus Laboratories, BMI-2104, Volume III, July 1984.

11. "Radionuclide Release Calculations for Selected Severe Accident Scenarios," Battelle's Columbus Division, NUREG/CR-4624, Vol. 4, July 1986.

12. "An Assessment of BWR Mark III Containment Challenges, Failure Modes, and Potential Improvements in Performance," NUREG/CR-5529, January 1991.

13. "The Response of BWR Mark III Containments to Short-term Station Blackout Severe Accident Sequences," NUREG/CR-5571, June 1991.

14. "Grand Gulf Nuclear Station - Integrated Containment Analysis," IDCOR Technical Report 23.1GG, March 1985.

15. "BWR Owners' Group - Emergency Procedure Guidelines, Revision 4," General Electric Co., NEDO-31331, March 1987

16. "Accident Management Information Needs," Idaho National Engineering Laboratory, NUREG/CR-5513, April 1990.

17. "Overview and Comparison of U.S. Commercial Nuclear Power Plants," NUREG/CR-5640, September 1990.

18. "Fission Product Release Characteristics into Containment Under Design Basis and Severe Accident Conditions," Brookhaven National Laboratory, NUREG/CR-4881, March 1988.

19. "Evaluation of Severe Accident Risks: Expert Opinion Elicitation on Source Term Uncertainty Issues," Rough Draft, Sandia National Laboratories, NUREG/CR-4551, Volume 2, July 1989.

20. "River Bend Station Updated Safety Analysis Report", Rev. 4, Gulf State Utilties Co., August 1991.

21. "Grand Gulf Emergency Operating Procedures", System Energy Resources, Inc., September 1988.

22. "Final Safety Analysis Report, Grand Gulf, Units 1", Mississippi Power & Light Co., November 1983.

23. "Perry Nuclear Power Plant, Updated Safety Analysis Report for Unit 1", Rev. ?, The Cleveland Electric Illuminating Co., March 1991.

24. "Radionuclide Release Under Specific LWR Accident Conditions," Battelle Columbus Laboratories, BMI-2104, Volume II, July 1984.

25. "Requirements for Emergency Response Capability," Supplement 1 to NUREG-0737, December 1982.

26. "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," NUREG-0654/FEMA-REP-1, Rev. 1, November 1980 and NUREG-0654/FEMA-REP-1, Rev. 1, Supp. 1, November 1987.

27. "Guidelines for the Preparation of Emergency Operating Procedures: Resolution of Comments on NUREG-0799," NUREG-0899, August 1982.

28. "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," USNRC Regulatory Guide 1.97, Revision 3, May 1983.

29. "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," USNRC Regulatory Guide 1.89, Revision 1, June 1984.

30. "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," The Institute of Electrical and Electronics Engineers, Inc., IEEE STD 323-1974.

31. "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors," American Nuclear Society, ANSI/ANS-4.5-1980, 1980.

32. "Identification and Assessment of Containment and Release Management Strategies for a BWR Mark I Containment," Brookhaven National Laboratory, NUREG/CR-5634, September 1991.

33. "GESSAR II, BWR/6 Nuclear Island Design," General Electric.

34. "Containment Performance Working Group Report," NUREG-1037, May 1985.

35. "Technical Aspects of Hydrogen Control and Combustion in Severe Light-Water Reactor Accidents," National Academy Press, 1987.

36. "Mark I Containment Performance Improvement Program," USNRC, SECY-89-017, January 1989.

37. "Completion of Containment Performance Improvement Program and Forwarding of Insights for use in the Individual Plant Examination for Severe Accident Vulnerabilities," U.S. NRC Generic Letter No. 88-20, Supplement No. 3, July 6, 1990.

38. "Functional Criteria for Emergency Response Facilities," NUREG-0696, February 1981.

39. "Platinum Catalytic Ignitors for Lean Hydrogen Air Mixture," Sandia National Laboratories, NUREG/CR-5086, September 1988.

40. "Application of Containment and Release Management to a PWR Ice-Condenser Plant," NUREG/CR-5707, July 1991.

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11. ABSTRACT (200 words or less)

This report identifies and assesses 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 BWR plant with a Mark III type of containment. Based on information available from probabilistic risk assessments and other existing severe accident research, and using simplified containment and release event trees, the report identifies the challenges a Mark III containment could face during the course of a severe accident, the mechanisms behind these challenges, and the strategies that could be used to mitigate the challenges. The strategies are linked to the general safety objectives which apply for containment and release management by means of a safety objective tree. The strategies were assessed by applying them to certain severe accident sequence categories deemed important for a Mark III containment because of one or more of the following characteristics: high probability of core damage, high consequences, lead to a number of challenges, and involve the failure of multiple systems.

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