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

Prepared by
C. C. Lin, J. R. Lehner

Brookhaven National Laboratory

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Abstract

Accident management strategies that have the potential to maintain containment integrity and control or mitigate the release of radioactivity following a severe accident at a boiling water reactor with a Mark II type of containment are identified and evaluated. The strategies are referred to as containment and release strategies. Using information available from probabilistic risk assessments and other existing severe accident research, and employing simplified containment and release event trees, this report identifies the challenges a Mark II containment may encounter during a severe accident, the mechanisms behind these challenges, and the strategies that could be used to mitigate the challenges. By means of a safety objective tree, the strategies are linked to the general safety objectives of containment and release management. As part of the assessment process, the strategies are applied to certain severe accident sequence categories deemed important to a Mark II containment. These sequence categories exhibit 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. The Limerick Generating Station is used as a representative Mark II plant to illustrate plant specifics in this report.

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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 II 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 Limerick Generating Station is used as the example plant in this report, but some of the variations among the other domestic Mark II plants are also discussed.

The present report emphasizes the use of existing plant capabilities for severe accident management. The containment and release management (CRM) strategies differ from the existing emergency procedure guidelines (EPG) primarily in terms of the conditions under which certain actions are undertaken and certain systems activated. For CRM, systems are often operated in an anticipatory instead of a response mode, and often beyond their design limits. Non-safety grade systems are also made use of for CRM. The plant features that are important to containment and release management of a BWR Mark II containment are reviewed to identify their function and performance under severe accident conditions. These include the containment design, the plant systems and the resources needed to support their operation, the emergency response facilities, the emergency procedure guidelines, 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 II plants in particular. Use was made of simplified containment and release event trees (CRET's) 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 concerns regarding operator action. The challenges to which a Mark II containment is subjected during a severe accident are in many ways similar to those faced by the other BWR containments, especially Mark I plants. Therefore many of the strategies are also similar. However, because of the different geometry of the Mark II containment, containment response, especially after vessel breach, can differ from that of a Mark I. The pedestal configuration of the reactor cavity area, which varies among Mark II plants, will influence the amount of core-concrete interactions taking place in the cavity and the extent to which corium will spread on the drywell floor. In plants where the corium can reach the downcomers, these may fail creating a bypass of the suppression pool. Even if corium does not reach the downcomers, suppression pool bypass can still occur due to failure of drains in the cavity region by corium attack or eventually by drywell floor failure. Besides increasing containment pressure loads, a bypass will significantly lessen the desirability of venting via the wetwell. Another consequence of these failures of the drywell to wetwell boundary is the high likelihood of steam explosions in the wetwell when corium drops into the suppression pool.

The BWR Emergency Procedure Guidelines, Revision 4 were used to estimate the operational response to a severe accident currently available at a Mark II 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 Limerick Generating Station. Often a single strategy would have multiple beneficial effects on accident management (e.g., drywell 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 effr.'

As is true for other containments, the lack of control room indications of containment variables in a Mark II could be a significant problem for accident management. This deficiency is particularly serious for a station blackout sequence. 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.

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
CAC	Containment Atmosphere Control
CCI	Core-Concrete Interaction
CDF	Core Damage Frequency
CET	Containment Event Tree
CF	Containment Failure
CHR	Containment Heat Removal
CM	Core Melt
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
DCH	Direct Containment Heating
DF	Decontamination Factor
DSIL	Drywell Spray Initiation Limit
DW	Drywell
ECCS	Emergency Core Cooling System
EOF	Emergency Operation Facility
EOP	Emergency Operating Procedures
EPG	Emergency Procedure Guidelines
ESW	Emergency Service Water
EVSE	Ex-Vessel Steam Explosion
FCI	Fuel-Coolant Interaction
FP	Fission Product
FW	Fire Water
GE	General Electric Company
HCLL	Heat Capacity Level Limit
HCTL	Heat Capacity Temperature Limit
HEPA	High Efficiency Particulate Air
HPCI	High Pressure Core Injection
HPCS	High Pressure Core Spray
HPME	High Pressure Melt Ejection
HPSW	High Pressure Service Water
HVAC	Heating, Ventilating and Air Conditioning

IAS	Instrument Air System
IDCOR	Industry Degraded Core Rulemaking
IEEE	Institute of Electrical and Electronics Engineering
ILRT	Integrated Leak Rate Testing
INEL	Idaho National Engineering Laboratory
IPE	Individual Plant Examinations
ISL	Interfacing Systems LOCA
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Core Injection
LPCS	Low Pressure Core Spray
MACCS	MELCOR Accident Consequence Code System
MSIV	Main Steam Isolation Valve
NMP2	Nine Mile Point Unit 2
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
OSC	Operational Support Center
PCIG	Primary Containment Instrument Gas
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
RERS	Reactor Enclosure Recirculation System
RHR	Residual Heat Removal
RHRSW	RHR Service Water
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup
SAM	Severe Accident Management
SARRP	Severe Accident Risk Reduction/Risk Rebaselining Program
SAS	Service Air System
SC	Secondary Containment
SBO	Station Blackout
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
SRV	Safety Relief Valve
STCP	Source Term Code Package

TAF	Top of Active Fuel
TPLL	Tail Pipe Level Limit
TSC	Technical Support Center
UHS	Ultimate Heat Sink
VB	Vessel Breach
WNP-2	WPPSS Nuclear Project No. 2
WPPSS	Washington Public Power Supply System
WW	Wetwell

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 preparation 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 additions, much can be accomplished through innovative use of already existing plant systems. Such an approach to risk reduction is preferable to one which relies on significant, and therefore costly, hardware changes or additions.

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 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. 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 [7-9].

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 II 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 an accident management plan.

The report can also furnish the reviewer of an accident management plan with a systematic overview of the challenges a Mark II 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 II 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 II 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 II 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 Sectic 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 II related safety studies. The sources used most frequently are the NUREG-1150 study and the supporting reports [4, 1(, the Mark II information produced for the NRC's SARRP [11] and CPI programs [12, 13], the containment and release management study for other BWR containments [7, 9] and a BNL study on severe accident mitigation strategies for a Mark I containment [14].

The BWR Owners' Group's Emergency Procedure Guidelines (EPGs), Revision 4 [15], were u ed as a baseline to gauge the guidance presently available to BWR Mark II 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 II 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 differ stages of the accident to check the plant status, as well as those they would need to implement the suggested strategies.

In the discussions which follow, when it is instructive to refer to specific plant features, the Limerick Generating Station is used as the exar Mark II plant.

2.2 Strategy Identification Process

Numerous sources, referenced throughout this report, were consulted to obtain information on the challenges a Mark II 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 II 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 II 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. For example, in the case of an interfacing systems LOCA (ISL), the radiological release phase will occur concurrently with the early phase. 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 II 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 II Containment System

The Mark II containment system includes a primary containment system and a secondary containment system. The primary containment system is a pressure suppression system. It consists of (1) a drywell, which has the shape of a truncated cone and houses the reactor vessel supported on a pedestal, (2) a cylindrical shaped pressure suppression chamber (wetwell), which is located directly below the drywell, separated from the drywell by a concrete diaphram slab (drywell floor), and which contains a large volume of water (suppression pool), (3) a downcomer vent system connecting the drywell and the suppression pool, (4) containment isolation systems, (5) containment heat removal systems, (6) combustible gas control systems, and (7) 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).

Enclosing the primary containment is the secondary containment. It consists of a reactor enclosure and a refueling area. The secondary containment provides housing for reactor auxiliary and service equipment, reactor refueling, and fuel storage facilities. It also retains airborne radioactive materials leaked from the primary containment in the event of an accident. This is achieved by a controlled, filtered and elevated release of the secondary containment atmosphere.

There are nine BWR facilities with Mark II containment designs in the United States. Two different BWR types are used in these facilities. The General Electric (GE) BWR/4 reactor design is used in five of the nine units and the BWR/5 design is used in the remaining four units. The major difference between the two BWR reactor designs is in the use of the high pressure emergency core cooling system (ECCS): the BWR/4 uses a turbine-driven high pressure coolant injection (HPCI) system and the BWR/5 uses a motor-driven high pressure core spray (HPCS) system. The motor-driven HPCS is also supported by a backup ac power from a dedicated diesel generator. Table 3.1 provides a listing of the domestic Mark II units, and the operator, the BWR type, and related electric power of each unit.

Figure 3.1 shows a schematic of the containment design for Limerick. The two parts of the primary containment, the drywell and the wetwell, comprise a structurally integrated concrete pressure vessel, lined with welded steel plate and provided with a steel domed head for closure at the top of the drywell. This construction is typical of all but one Mark II containment. Instead of a concrete structure, WNP-2 utilizes a free-standing steel primary containment, surrounded by a reinforced concrete structure providing support and biological shielding.

Table 3.2 ... the values of some plant parameters for the nine Mark II units located at six plant sites. As shown in Table 3 ... rated thermal power of the eight operating units (except Shoreham) is within a narrow range, varying from 3,293 to 3,448 MWt, the drywell free volume varies from 200,500 to 303,400 ft^3, the wetwell free volume varies from 144,200 to 192,000 ft^3, and the suppression pool water volume varies from 112,200 to 154,800 ft^3. Also presented in Table 3.2 are the relative elevation of the floor inside the reactor pedestal (in-pedestal) region to the drywell floor, the number of downcomers in the in-pedestal region, and the design pressure and temperature of the containment. The characteristics of the in-pedestal design is important in severe accident progression and will be discussed in more detail later in this section.

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, the suppression pool, and the in-pedestal design. In the following discussions Limerick is used as the representative Mark II plant. It is important to note that there are variations among the operating Mark II 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. One example of important design variations is the design of the in-pedestal region. Because of its importance in severe accident progression, the in-pedestal design of the various Mark II containments is discussed in some detail in this report.

3.1.1 Primary Containment

The primary containment for Limerick has an internal design pressure of 55 psig and an external design pressure of 5 psid. The atmospheric design temperature is 340 °F for the drywell and 220 °F for the wetwell. The leakage rate of the primary containment is limited to less than 0.5% free volume per day at design pressure and temperature. To reduce the possibility of hydrogen combustion, the primary containment is maintained in an inerted state by the operation of a nitrogen inerting system, which is a part of the containment atmospheric control (CAC) system of Limerick. Containment inerting is achieved by maintaining a nitrogen rich containment atmosphere whose oxygen concentration is less than 4%.

Because of the leaktight design of the primary containment, release of fission products to the environment is insignificant if the primary containment remains intact and is not bypassed. If the containment does fail, the consequence of fission product release will depend strongly on the time and mode of containment failure. A larger failure size will result in a more rapid discharge, less residence time for natural deposition, and consequently, in most cases a greater release of radioactive materials to the environment. A failure in the wetwell airspace (without suppression pool bypass) will reduce fission product release to the environment because the fission products will be scrubbed by passing through the suppression pool. 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.

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. Detailed discussions of these issues are presented below.

3.1.1.1 Containment Pressure Capability and Failure Mode

Although the design containment pressure of Limerick is 55 psig, the actual containment failure pressure is expected to be much higher. The containment pressure capability for Limerick has been estimated to be between 120 to 170 psig [17] at normal temperature. Based on these estimates, the Containment Performance Working Group Report concluded that a pressure capability of 140 psig was an acceptable upper limit of the internal pressure. In analyses of the Severe Accident Risk Reduction/Risk Rebaselining Program (SARRP) containment failure by overpressurization was assumed to occur at a pressure of 145 psia with a break area of 7 ft^2 in the drywell [11].

There is considerable uncertainty in estimating containment strength and failure mode. Probabilistic descriptions of containment failure pressure and failure mode were used in the NUREG-1150 studies of both the Mark I and the Mark III containments [4]. (The Mark II containment is not included in the NUREG-1150 information published to date [4].) The containment strength and failure mode may also depend on containment temperature, and containment leakage may develop at large penetrations before containment failure pressure is reached [17]. Since containment temperature of over 1,000°F has been predicted in some severe accident analyses [13,14], containment strength and material properties may be degraded during these accidents. Available data to date indicates that a likely failure location due to a combination of high containment temperature and pressure is the drywell head flange seal. It is a serious failure mode, because release through the drywell head bypasses both the suppression pool and a large part of the reactor building. The fission products are therfore not scrubbed before they are released to the environment.

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. Under the assumptions of the modelling for severe accidents used in the Source Term Code Package (STCP), 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.

3.1.1.3 In-Pedestal Design

The design of the region inside the reactor pedestal significantly influences the progression of a severe accident after the debris is discharged onto the drywell floor. The design features that are most important to accident progression are the relative elevation of the in-pedestal floor to the drywell floor and the existence of downcomers inside the pedestal region. Figure 3.2 shows the various in-pedestal design of the domestic Mark II containments. In general, the BWR/5 plants have a recessed in-pedestal region (reactor cavity) and the BWR/4 plants have a flat in-pedestal floor at approximately the same elevation as the ex-pedestal drywell floor (see Table 3.2). Among the domestic Mark II plants, only Nine Mile Point Unit 2 (NMP2) and Shoreham have downcomers inside the pedestal region.

After vessel failure and the discharge of core debris, a ressessed cavity would confine the core debris (corium) in the cavity. Extensive corium-concrete interaction (CCI) is expected to occur because the potential for corium cooling is minimal. On the other hand, a shallow reactor cavity would allow the corium to spread out through the personnel pathway onto the drywell floor. A portion of the corium could enter the first row of downcomer pipes. The remaining portion would be cooled by heat losses to the containment atmosphere and the drywell floor [20], and by the drywell spray if it is operational.

For plants that have downcomers in the pedestal region, corium released from the vessel would enter the suppression pool rapidly. this design may eliminate the problems associated with CCI, if the corium is primarily in liquid phase and the vessel is not pressurized, but increases the potential of a severe and damaging fuel-coolant interaction (FCI, or steam explosion).

The potential for a steam explosion as corium flows down the downcomer pipes into the suppression pool has been discussed in Reference 20. The thermal attack by corium could also fail the downcomer pipes and cause a suppression pool bypass. Suppression pool bypass could have a significant impact on containment integrity and, therefore, fission product release.

In addition to the downcomer pipes, there are drain tubes located in the drywell floor. (All plants, except Susquehanna, have in-pedestal drains into the wetwell.) These drain tubes could also fail by corium attack. This would result in a suppression pool bypass and FCI when the corium falls into the suppression pool through the failed drain tubes. Since there are only a few drain tubes and their size is much smaller than the downcomer pipes, the drain tubes are expected to have a smaller impact on containment loading than downcomers [14].

3.1.2 Suppression Pool

The suppression pool (SP) is designed to condense the steam from the reactor pressure vessel (RPV) during a postulated LOCA event. It is connected to the drywell through a downcomer vent system (Figure 3.1). These downcomers are 24 inches in diameter and terminate about 11 ft below the normal water level of the suppression pool (Limerick). Typically there are between 82 to 129 downcomers in a Mark II containment (Limerick has 87 downcomers).

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 downcomer vents 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 differential pressure if the wetwell pressure exceeds that of the drywell. For Limerick there are four pairs of 24-inch vacuum relief valves (two valves in each pair are mounted in series) installed in the wetwell and attached to the downcomer pipes above the suppression pool water level. The set pressure of these vaccum breakers is 0.5 psid.

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 HPCI for BWR/4, or RCIC and HPCS for BWR/5), and the principal water source for the low pressure ECCS systems (LPCS and LPCI) and the containment spray (CS) systems. LPCI and CS are different operating modes of the RHR system and, as such, share components of the RHR system[1].

The energy deposited into the suppression pool during an accident can be removed via the RHR heat exchangers. The ultimate heat sink for the RHR heat exchangers is provided by the RHR service water (RHRSW) 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 the pool 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 (FP) produced in-vessel and released through the SRV spargers. The decontamination factor (DF) used in the NUREG-1150 analysis for in-vessel releases of a Mark I containment (Peach Bottom) ranges from 1.2 to 4000 with a median value of 80 [4]. In comparison, the DF for ex-vessel releases and flows passing through the downcomer vents is smaller. The DF values used in the NUREG-1150 analysis for the same Mark I plant range from 1 to 90, with a median value of 10.

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[1]In Limerick a cross connection line exists between the RHR service water system and one of the RHR loops, and this makes the RHRSW available for the RHR system.

Since SRV discharge line and downcomer arrangements for a Mark II containment are similar to those of a Mark I containment, the suppression pool decontamination factors are expected to be similar. In fact, the DFs in a Mark II containment are expected to be greater because of the greater submergences of both the SRV discharge lines and downcomers in a Mark II containment.

After the RPV is breached, fission products are released to the drywell. Part of these releases will pass through the downcomer vents and be scrubbed by the suppression pool. The fission products that remain in the drywell atmosphere will discharge directly to the reactor building without suppression pool scrubbing if the containment failed in the drywell. 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. A saturated suppression pool may cause the pumps that take suction from the suppression pool to fail from cavitation[2]. This is more likely to happen immediately after containment failure or venting, when the containment atmosphere is rapidly depressurized and the SP could flash. It is therefore important to switch these pumps to an alternate water source before such a detrimental condition develops. 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 one of the source term issues addressed by expert elicitation in NUREG-1150 because of its uncertainty and importance.

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 plant specific emergency operating procedures provide specific directions and procedures to control suppression pool water level in an accident. Water can be added to the suppression pool if water sources external to the containment are used for the ECCS (e.g., condensate storage tank, CST) or the RHR system (e.g., RHRSW). Water can also be removed from the suppression pool to the

[2]At the BWR/5 plants and Limerick, the RHR pumps can pump saturated water without failure [12].

water tanks outside the containment through test lines for the ECCS or RHR system (e.g., CST). 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. In Limerick, the reactor enclosure recirculation system (RERS) and the standby gas treatment system (SGTS) are 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 is made up of the reactor enclosure (or reactor building) and a refueling bay area. At multi-unit sites, a single secondary containment is divided into distinct isolatable zones. For Limerick, a two-unit site, there are three zones. Zones I and II are the Unit 1 and Unit II reactor enclosures. Zone III is the common refueling area.

The construction of the secondary containment is similar in all Mark II plants. The lower levels of the secondary containment are reinforced concrete structures. Above this, the building structure consists of metal siding supported on a steel superstructure (Limerick uses a reinforced concrete superstructure). The roof is usually constructed of steel decking (Limerick uses a reinforced concrete slab). The internal design pressure of the secondary containment is usually 0.25 psi and the design leakage rate is about 100% free volume per day at 0.25 inches water pressure. If the internal pressure exceeds the design pressure, the excess pressure is vented to the atmosphere through blow-out panels located in the superstructure of the buildings. The ultimate failure pressure of the secondary containment is plant specific. However, in general, it cannot take a significant internal pressure load.

The secondary containment houses equipment important to plant operation and accident management, e.g., the ECCS and RHR system pumps. The reactor building heating and ventilating system is designed to provide suitable environmental conditions for personnel and equipment. The system is isolated upon receipt of a plant isolation signal. The same signal also actuates the standby gas treatment system (SGTS), which is designed to limit the ground level release from the reactor building by providing (1) a filtered release of the reactor building atmosphere removing radioactive particulates and halogens, and (2) an elevated release via a vent or a stack. The height of the release point is 200 ft above ground level for Limerick and varies from about 200 to 430 ft for other Mark II plants (Table 3.2).

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

3.1.3.1 Secondary Containment Decontamination Factors

The secondary containment provides additional fission product retention from natural processes such as aerosol deposition and vapor condensation. The decontamination factor of the reactor building is primarily a function of the residence time and thermal hydraulics of the transporting gases in the building and thus depends on (1) the size and location of the primary containment break, (2) the internal design of the secondary containment (e.g., compartmentalization), (3) the ability of the reactor building to remain intact, (4) the magnitude and frequency of hydrogen burns, and (5) the driving force from the primary containment. The reactor building decontamination factors used in the NUREG-1150 analysis for a Mark I plant range from 1.1 to 10 with a median value of three for typical accident conditions [4].

There are significant uncertainties in estimating the effectiveness of secondary containment decontamination. This is reflected in three issues considered by the expert panels in the NUREG-1150 analysis. They are (1) the strength of the reactor building, (2) the probability and effects of hydrogen combustion in the reactor building, and (3) the reactor building decontamination factor.

3.1.3.2 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 fuel handling area or the reactor enclosure under accident conditions. It provides a filtered and controlled release of the secondary containment atmosphere. 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 Limerick is presented below.

The SGTS consists of two full capacity exhaust fans (each with a controllable capacity of from 500 to 3,000 cfm), two full capacity filter trains, and two redundant sets of the associated ductwork, dampers, and controls. Each filter train consists of an electric air heater, two banks of HEPA filters (upstream and downstream of charcoal adsorber), and a vertical 8-inch deep charcoal adsorber bed. The HEPA filters can remove 99.9% of all particles greater than 0.3 microns in diameter. Iodine is removed by activated charcoal beds, which can remove more than 99% of elemental iodine and 99% of methyliodide at 70% relative humidity. The maximum loading of the charcoal bed for Limerick is about six pounds (based on a maximum loading of 2.5 milligrams of iodine per gram of activated charcoal and 2,400 pounds of charcoal [21]). The charcoal has an ignition temperature of greater than 626°F. A water flooding system is provided within the charcoal bed for fire protection. The presence of adsorbed water on the charcoal surface will substantially affect its efficiency by reducing the surface area available for the trapping of volatile forms of radioactive iodine.

The SGTS can take suction from the RERS or the containment purge system. Flow is initiated from the RERS automatically upon receipt of a secondary containment isolation signal (a safety-related mode of SGTS operation). The RERS recirculates the air in the secondary containment and reduces the activity released through the SGTS. The RERS consists of two full capacity recirculation fans (each with a capacity of 60,000 cfm) and two full filter trains with HEPA filters and charcoal adsorbers, similar to that of the SGTS. The RERS serves as the initial cleanup system and the SGTS serves as the final cleanup system for the gases discharged from the reactor enclosure.

Upon receipt of a secondary containment isolation signal, both SGTS trains will start automatically. Following the initial start, the operators may elect to place one of the SGTS trains in a standby position. The SGTS is manually actuated for its non-safety related function of reducing halogen and particulate concentrations in gases purged from the primary containment.

The SGTS is designed for the amount of aerosols expected after a postulated loss of coolant accident (LOCA) event or a fuel handling accident. The amount of aerosols released during a severe accident may be much greater. 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 [22]. 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 fans 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. Limerick plant parameters are used for illustration.

3.2.1 Primary Containment Ventilation, Cooling, and Water Supply

In Limerick, ventilation and cooling of the primary containment is normally provided by two systems: the containment atmospheric control (CAC) system and the drywell air cooling system. The RHR system, used for SP cooling during normal operation, is used during an accident for emergency cooling of the primary containment. A short description of these systems 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 CAC system of Limerick incorporates features for accomplishing a number of functions, including inerting of the primary containment with nitrogen, purging of the primary containment, limiting the differential pressure between drywell and wetwell, monitoring of hydrogen and oxygen concentrations in the primary containment, and controlling combustible gas concentration in the primary containment after LOCA. The nitrogen inerting part of the CAC system consists of two liquid nitrogen storage tanks and one steam-heated water bath vaporizer. It provides nitrogen to the containment and has a normal pressure of 40 psig [23]. The containment is inerted by high-volume purging during its normal inerting and de-inerting operations, e.g., during normal startup or shutdown. There are four high-volume purge lines: a 24-inch drywell purge supply line, a 24-inch drywell purge exhaust line, a 20-inch wetwell purge supply line, and an 18-inch wetwell purge exhaust line. Gases from high-volume purging are processed by the SGTS prior to release to the environment. Low-volume purging is used during reactor operation to maintain the pressure and oxygen concentration of the primary containment within specified ranges. It is also used for post-LOCA oxygen concentration control as a backup to the hydrogen recombiner system. Low-volume purging uses a 1-inch supply line and a 2-inch exhaust line. All of these purge lines can also be used for containment venting, which is one of the severe accident strategies discussed later in this report.

The drywell air cooling system serves to remove heat from the drywell during normal plant operations and to maintain air circulation in the drywell under accident conditions. It is designed to limit the temperature inside the drywell to 135°F during normal operation and to maintain the drywell atmosphere in a thoroughly mixed condition following an accident to prevent stratification of oxygen in the drywell. The drywell air cooling system includes eight drywell unit coolers, each of which contains two redundant cooling coils and two redundant fans. The flow rate of each unit cooler is 7,000 cfm and the cooling capacity of each cooling coil is 0.575 MBtu/hr. Chilled water is supplied to the unit coolers by the drywell chilled water system during normal operation and by the reactor enclosure cooling water system during the loss of offsite power, when the chilled water is not available.

The RHR system, in its containment cooling operation, is used to prevent excessive containment temperature following a LOCA so that containment integrity is maintained. The RHR system in Limerick is comprised of four independent loops. Each loop contains a motor-driven pump, piping, valves, instrumentation, and controls. The RHR pumps take suction from the suppression pool and are powered by the emergency diesel generators if offsite power is not available. Two of the loops have heat exchangers that are cooled by the RHR service water (RHRSW). The heat exchanger in each loop is estimated to have a heat removal capacity of approximately 122 MBtu/hr (based on a 95°F service water temperature and a 212°F suppression pool temperature). The combined heat removal capacity of both RHR heat exchangers is about 2.2% of the rated thermal power of the reactor. The RHR heat removal capacity may be higher if the pool temperature is higher than 212°F, but corrosion or biofouling could also reduce this capacity significantly. The RHR system can be operated in either the suppression pool cooling (SPC) mode or the containment spray (CS) mode for containment cooling. Both of these modes are manually actuated. Since they share systems with the RHR core injection mode, their use is prohibited by an interlock, unless the core has been reflooded to two-thirds the core height. The control room operator can override the interlock using a keylock.

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 Limerick, a crosstie with the RHR service water (RHRSW) system is already available. The Limerick RHRSW system takes suction from the spray pond, the plant's ultimate heat sink (UHS). The UHS is designed to provide cooling water, and act as a heat sink, for the emergency service water (ESW) system and the RHRSW system during accident conditions. The spray pond has a storage volume of close to 30 million gallons, and makeup water is available from the Schuylkill river. The RHRSW system automatically aligns itself to the spray pond mode upon standby diesel start, and can be manually aligned to a cooling tower, which cools and recirculates the RHRSW, if this mode is available.

A crosstie of the RHR system can also be made with the fire water (FW) system. The fire water system in Limerick has a diesel-driven pump as backup 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. Each of the two pumps can provide a flow capacity of 2500 gpm at 125 psig, and is capable of taking suction from either of the two 7.2 million gallon cooling tower basins of the two units.

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 Limerick 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 Limerick station has two independent sources of offsite power. In the event of the loss of any one of the two connected offsite power sources, a third independent offsite source can be connected for emergency use to supply the engineered safeguard loads . The onsite standby ac power is supplied by four independent diesel generators (eight for both units). Each diesel generator is exclusively connected to one of the four independent standby power divisions. The diesel generators start automatically on a total loss of offsite power. 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. However, some of these trip signals are bypassed when the diesel generator receives an emergency (LOCA) start signal [21]. For a BWR/5 plant there is an additional diesel generator dedicated to the HPCS system.

The dc power system consists of independent Class IE and non-Class IE dc power systems. In Limerick, there are four independent divisions of Class IE dc systems for each unit: two 125/250 V, three-wire systems; and two 125 V, two-wire systems. Each 125/250 V system is comprised of two 125-V batteries, each with its own charger, a fuse box for protection of each of the several 125 V power distribution circuits supplying 125/250 V motor control centers, and two 125 V power distribution panels. (The 250 V dc power is used to supply power for the larger loads, such as dc motor-driven pumps and valves. It is supplied by the two 125 V sources of the system, connected in series and distributed through 250 V dc motor control centers.) Each 125 V system is comprised of one 125 V battery with its own charger, a fuse box, and two 125 V power distribution panels. The non-Class IE dc systems for Limerick consist of a 250 V non-Class IE dc system and a 125/250 V non-Class IE system. 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 Limerick, the Class IE batteries have sufficient capacity to supply the required loads for four hours in a station blackout (SBO) event [2].

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. The plant dc power can also be extended by utilizing the non-class IE dc systems. Detailed discussions of these strategies related to loss of power can be found in NUREG/CR-5474.

The pneumatic supplies in Limerick are provided by the instrument air, the service air, and primary containment instrument gas (PCIG) systems. The instrument air system (IAS) provides filtered dry, oil free, compressed air for air operated control devices and instruments throughout the plant. The service air system (SAS) is used to provide filtered compressed air for service and maintenance operations and to provide a backup to the instrument air system. The PCIG system provides a supply of compressed nitrogen gas for operating the pneumatic devices located in the containment. The IAS backs up the PCIG system through two control room operated valves.

The instrument air system for Limerick consists of two full-capacity compressors, complete with filter, air dryer, and aftercooler. During normal operation, one of the instrument air compressors is selected as the lead compressor, while the other serves as a standby. Each IAS compressor has a capacity of 397 scfm and can deliver compressed air at 110 psig to support the operation of safety related equipment. The air receiver of each IAS train has a capacity of 223 ft^3. The service air system consists of one full-capacity compressor with a capacity of 397 scfm and a pressure rating of 110 psig. 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. There is a single backup service air system that can supply either of the two Limerick units in the case of loss of service air.

The PCIG system of Limerick consists of two full-capacity trains of gas filters, compressors, aftercoolers, moisture separators, dryers, receivers, and associated piping, valves, controls, and instrumentation. The PCIG compressors take suction from the drywell, and each has a capacity of 10 scfm and a pressure rating of 110 psig. The gas receiver of each PCIG train has a capacity of 80 ft^3 and the two trains are cross-connected by a common header. A backup to the PCIG system is provided by an intertie to the IAS via a normally closed valve remote-manually operated from the control room. Vital components, such as MSIVs and SRVs, are provided with accumulators to assure reliable function without compressor operation. Some plants, such as Limerick, also utilize a long-term, backup, safety-related, pneumatic supply to the ADS valve accumulators. In Limerick, gas bottles with seismic Category I supports are provided for operation of the ADS valves for seven days and a seismic Category I external connection is provided outside the reactor enclosure for the operation of the ADS valves beyond seven days.

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 four RHR loops can be utilized by the CS system. Each of the two loops forms a completely independent and redundant CS train containing its own motor-operated valves, motor-driven pump, heat exchangers, drywell spray header. The wetwell spray ring is common to both loops. The CS system normally takes suction from the suppression pool and each of the two CS loops can deliver a flow rate of 10,000 gpm to the containment, 95-100% of this flow can be delivered to the drywell spray header with the rest going to the wetwell spray ring. The capability to use the RHRSW system as a CS water source is also available via an existing crosstie (see Section 3.2.1).

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].

There are possible adverse effects associated with the operation of the CS system, particularly after the containment has been vented. 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. A successful implementation of a containment venting strategy requires: (1) establishing an optimum venting pressure, (2) identifying and prioritizing available vent paths, (3) evaluating the flow capacity of the identified vent paths, (4) assessing the structural capability and loading of the paths during venting, (5) appraising potential adverse effects, (6) investigating the operability of the vent paths under severe accident conditions, and (7) preparing containment venting guidelines or procedures. Some of the above issues are discussed in the following. More detailed discussions of these issues and possible hardware modifications to improve containment venting have been presented in the Mark I report on CRM [7].

The objective of containment venting in the BWR EPGs is to prevent containment 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 pressu 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, and (3) the maximum containment pressure at which SRVs can be opened and will remain opened. 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 II containment is tested to 1.15 times the design pressure (63 psig for Limerick) 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. In Limerick, 70 psig was selected as the PCPL [14]. This is about 1.3 times the design pressure, greater than its structural integrity test pressure (1.15 times design pressure).

The venting area required to maintain the containment pressure below the PCPL depends on mass and energy input rate to the containment atmosphere, which depends on the type of accident sequence that is occurring and the progression of the accident. The selection of the vent paths is usually limited to existing primary containment penetrations. In Limerick, the penetration lines of the containment purge systems (Section 3.2.1) can be used for containment venting. Other lines that can be used for containment venting include those associated with the integrated leak rate testing (ILRT, two 6-inch lines) and drywell sump drains (two 4-inch lines). The flow capacities of the selected vent paths can be evaluated against predicted energy input rates to assure that they are sufficient for successful venting. The evaluation can also provide information for selecting the preferred vent paths for a particular accident sequence and accident progression conditions, if these are known. For example, an ATWS event requires a large venting area in a short duration. In this event it may not be prudent to spend time to open small vent paths first and the above information will help determine the preferred vent paths to be opened.

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, secondary containment contamination and the resulting loss of safety related equipment or loss of accessibility for operator actions, and fission product releases to the environment. Because of the above potential adverse effects, venting may not lead to a desirable result. For example, using pessimistic assumptions regarding reactor coolant injection failure and secondary containment bypass, the radioactivity release at Shoreham was predicted to be higher for the venting case than the no-venting case [24]. The effect of containment venting for a Mark II containment is also affected by the likelihood of downcomer or drain line failure (by the attack of molten core debris) and the resultant loss of suppression pool scrubbing [25].

The installation of an improved hardened vent capability for a BWR plant to remove some of the adverse effects has been discussed by the NRC in the containment performance improvement (CPI) program [25,26]. However, less definitive conclusions have been reached regarding the need for improved venting for a Mark II containment than for a Mark I containment. The risk reduction to be gained from improvements to the vent system for a Mark II plant maybe less than that for a Mark I plant. As a result, the NRC staff has recommended that venting be evaluated as part of the Individual Plant Examinations (IPE) process for a Mark II plant. Each mark II plant would use its own plant specific hardware and procedures to determine how best to maximize the benefit from venting and minimize potential adverse effects [26].

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, (3) the time and manpower available to perform the required venting operations, (4) the design and environmental qualification of the equipment, and (5) accessibility to needed venting equipment if local operation is required. 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 [27] and NUREG-0654 regarding radiological emergency response plans and preparedness [28]. 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 [28]. 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.

Plant Capabilities

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 [29]. 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 a Mark II containment are (1) high suppression pool temperature (e.g., above 95 °F), (2) high drywell temperature (e.g., above 135°F), (3) high drywell pressure (e.g., above 2 psig), (4) high or low suppression pool water level, or (5) high containment hydrogen concentration (e.g., greater than 1%) [15]. 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 hydrogen and oxygen concentrations in 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 shut down 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 pool swell 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 shut down 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 assumptions 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 for severe accident conditions 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) directi[illegible]ctions 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 enviro[illegible] during and following an accident is described in Regulatory Guide 1.97 (Rev. 3) [30]. 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. The five types are: Type A, those variables that provide primary information needed to permit the control room operating personnel to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events; Type B, those variables that provide information to indicate whether plant safety functions are being accomplished; Type C, those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product release; Type D, those variables that provide information to indicate the operation of individual safety systems and other systems important to safety, and Type E, those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and for continuously assessing such releases [30].

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 [27]. 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 (for example, using local instrument taps) may be of benefit.

The three qualification categories referred to above 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 [31]. 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 [32] are acceptable to Regulatory Guide 1.89 [31]. 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 [33] 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 Limerick, 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 recommended by CPI for RPV depressurization may also be desirable.

Table 3.1 Domestic BWR Facilities With the Mark II Containment System

Plants Licensed for BWR Operation	Licensee	Net MWe/Un.	BWR Type
LaSalle Units 1 and 2	Commonwealth Edison	1,026	5
Limerick Units 1 and 2	Philadelphia Electric Co.	1,055	4
Nine Mile Point 2	Niagara Mohawk Power Corp.	1,072	5
Shoreham*	Long Island Lighting Co.	809	4
Susquehanna Units 1 and 2	Pennsylvania Power & Light Co.	1,035	4
WNP-2	Washington Public Power Supply System	1,095	5

*The Shoreham unit received a full power operating license on April 20, 1989. It achieved criticality and produced power, but closed before it could begin commercial operation by agreement between the Long Island Lighting Company and New York State.

Table 3.2 Comparison of Rated Power and Containment Design Characteristics of Mark II Plants

					In-Pedestal Region		Design Pressure		Design Temperature (°F)		
	Rated Thermal Power (MWt)	Drywell Free Volume (ft^3)	Wetwell Free Volume (ft^3)	Suppression Pool Water Volume (ft^3)	Elevation Relative to DW Floor	Number of Downcomers	Internal (psig)	External (psid)	Drywell	Wetwell	Height of Elevated Release Point (ft. above ground)
LaSalle 1 & 2	3,434	221,500	166,400	124,000	Below	0	45	-5	340	275	370
Limerick 1 & 2	3,293	248,700	135,400	124,700	Same	0	55	-5	340	220	200
Nine Mile Point 2	3,463	203,400	192,000	154,800	Below	8	45	-5	340	275	
Shoreham	2,436	192,500	134,000	81,400	Same	4	48	-5	300	225	230
Susquehanna 1 & 2	3,293	239,600	153,800	127,000	Same	0	53	-5	340	220	200
WNP-2	3,468	200,500	144,200	112,200	Below	0	45	-5	340	275	230

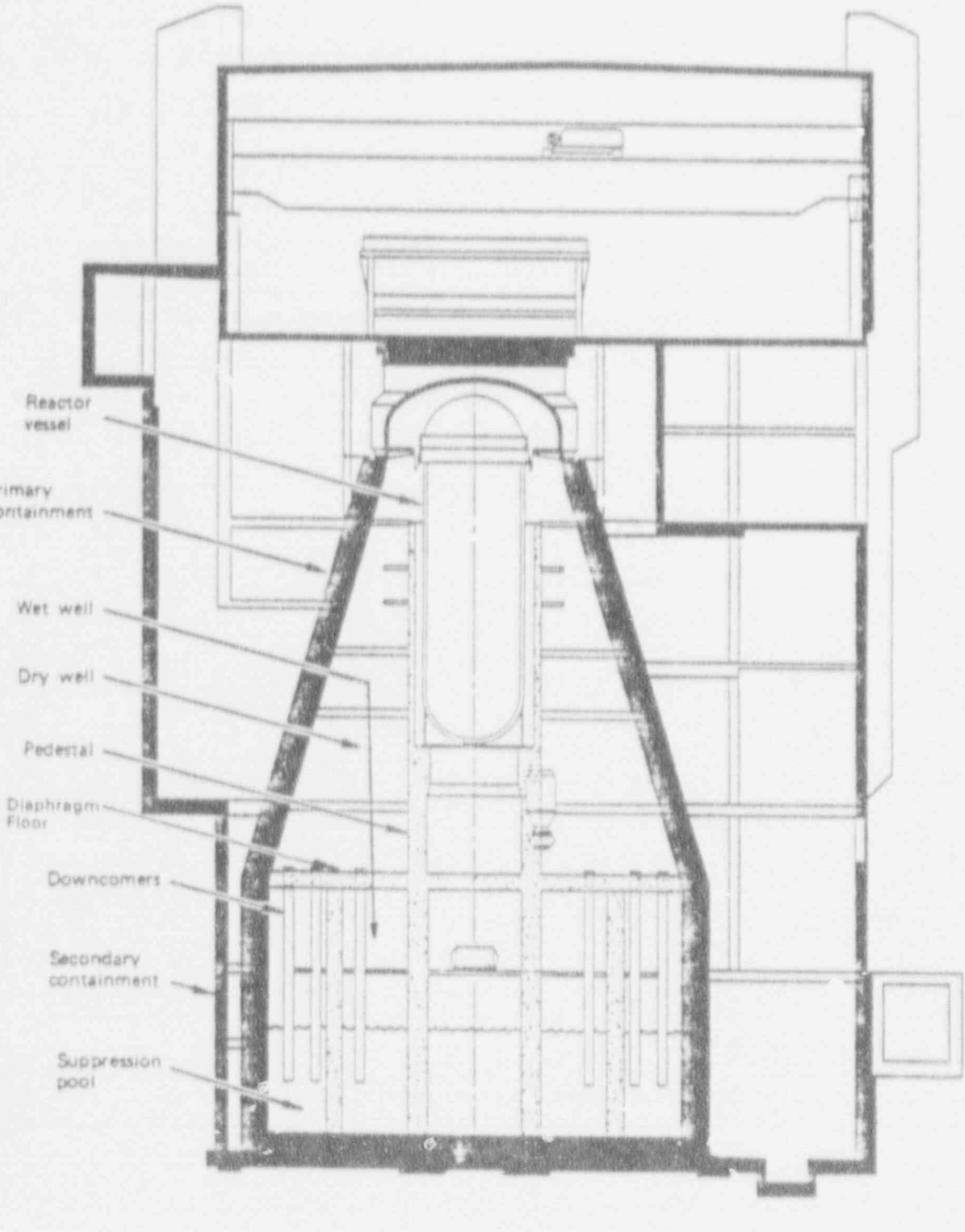

Figure 3.1 Limerick Containment

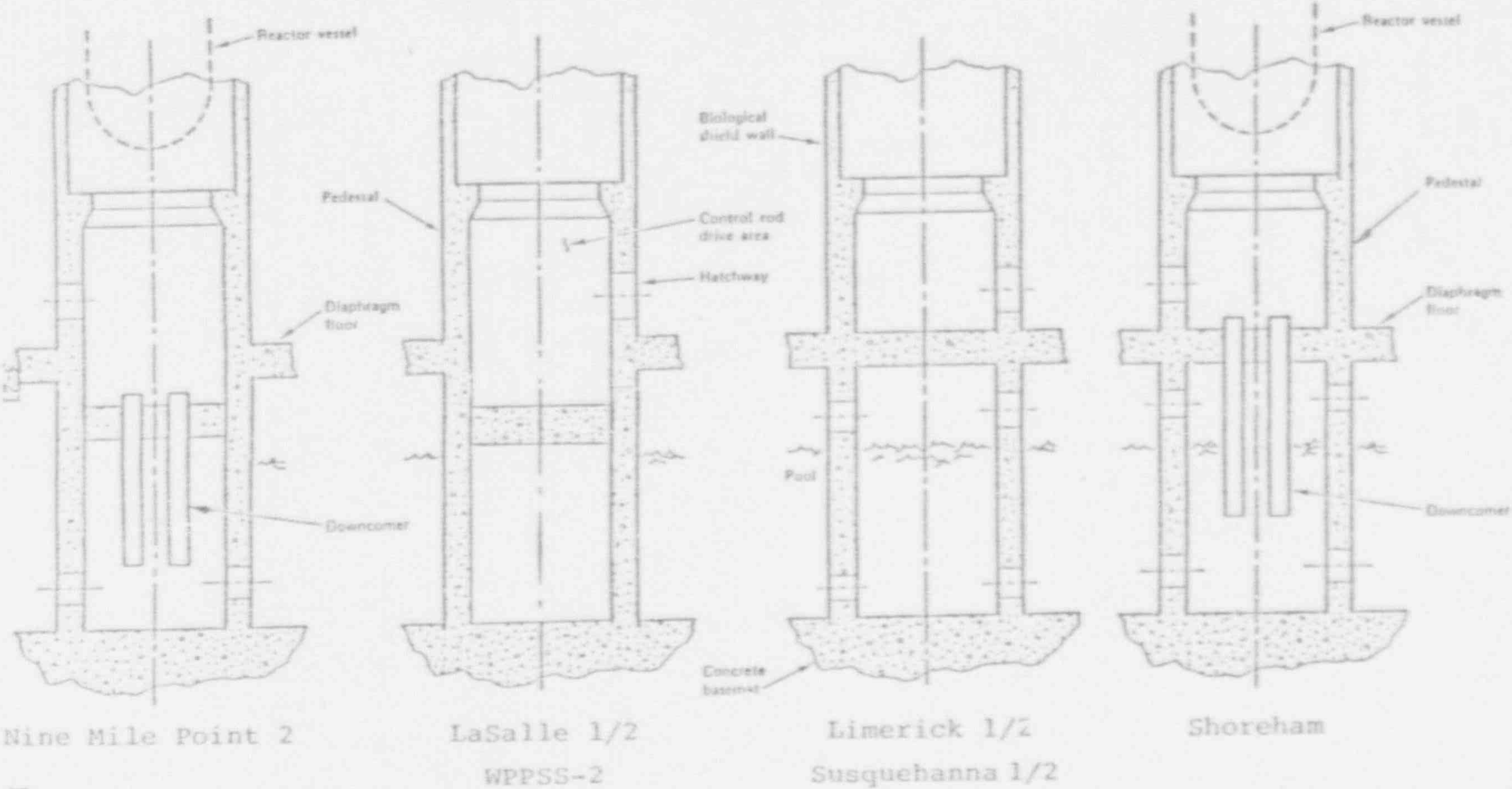

Figure 3.2 Variations in the Mark II Pedestal Configuration

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 [7]. Existing information on severe accidents is reviewed to identify (1) the challenges a Mark II 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 [7].) 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 4.2).

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 [7].

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. Most recent information on containment event trees (CET), or accident progression event trees (APET), is available in the NUREG-1150 report and its supporting documents. Since the Mark II containment is not one of the containment designs evaluated in NUREG-1150 [4], such information for a Mark II containment is not provided in Reference 4. However, similar information for Mark II containments can be found in the documents on probabilistic risk assessments (PRAs) prepared by the utilities [34-37], review of these PRAs by NRC contractors [38, 39], and studies associated with NRC's CPI program [12,13].

The data provided in the NUREG-1150 documents, although not specifically prepared for a Mark II containment, can still be useful in the evaluation of accident progression in a Mark II containment. Table 4.1 presents the values of some the plant parameters for a Mark II containment (Limerick) and other BWR containment types that are included in the NUREG-1150 study (Peach Bottom for Mark I and Grand Gulf for Mark III). Table 4.1 shows significant similarity between a Mark I containment and a Mark II containment. However, there are differences that may affect the progression of a severe accident. The containment free volume of a Mark II containment is greater than that of a Mark I containment (by about 30-40%). The containment pressure rise for a Mark II containment from mass and energy addition during a severe accident may therefore be slower than that for a Mark I containment. This results in a longer duration to containment overpressure failure, and allows additional time to plant personnel for containment venting and emergency response operations.

In addition to the parameters presented in Table 4.1, the relative configuration of the drywell, the suppression pool, and the wetwell is also important to accident progression. For both Mark I and Mark II containments, the drywell and the wetwell are separate compartments connected by a vertical downcomer vent system. This is in contrast to a Mark III containment, where the wetwell encloses the drywell, and a horizontal vent system connects these two volumes. The consequence of a drywell failure is therefore similar between a Mark I and a Mark II containment, but quite different for a Mark III containment. However, despite the general similarity between a Mark I and a Mark II containment, there are specific features that are different in these two containment types which have important effects on the progression and consequence of a severe accident. The smaller drywell floor area and the steel shell construction for the Mark I containment make it more liable to drywell shell melt through (by the attack of the hot core debris discharged from the RPV after vessel breach). On the other hand the location of the downcomers and the design of the in-pedestal region for the Mark II containment make it more liable to a suppression pool bypass after vessel breach (Section 3.1.1.3). The melt through of the drywell floor by the corium is also more likely for a Mark II containment than a Mark I containment because of the smaller drywell floor thickness of the Mark II containment. While a drywell floor failure for a Mark II containment allows the corium to fall into the suppression pool, resulting in a suppression pool bypass and fuel coolant interaction (FCI), a similar failure for a Mark I containment will result in a breach of the containment and fission product release to the environment.

The important sequences that can lead to a severe accident (or plant damage states, PDSs) include station blackout (SBO), anticipated transient without scram (ATWS), transients, loss of coolant accident (LOCA), and loss of containment heat removal (TW sequence). Table 4.2 shows the contributions from these PDSs to the total plant core damage frequency (CDF) for Limerick (Mark II), Peach Bottom (Mark I), Grand Gulf, and a generic Mark II containment used in the CPI program (based on a BWR/4 plant). For the plants presented in Table 4.2, both Limerick and Peach Bottom utilize a BWR/4 reactor design and use a turbine-driven HPCI system for the high pressure ECCS. Grand Gulf utilizes a BWR/6 reactor design and uses a motor-driven HPCS system with a dedicated diesel generator for the high pressure ECCS, a feature similar to those Mark II plants utilizing a BWR/5 reactor design. In general, SBO is the largest contributor to the total CDF for all BWR containment types. Transients, as shown in Table 4.2, are a large contributor to the CDF of a Mark II containment. The leading transient sequence is normally the one where the high pressure injection is lost and the low pressure injection is not available due to a RPV depressurization failure (a TQUX sequence). The large contribution of transient sequences to the total CDF of a Mark II containment is partly because of the use of a higher ADS failure probability (based on the ADS initiation logic prior to the recommended modification by Reference 40). Another important sequence shown in Table 4.2 is the ATWS sequence. It is a significant contributor to the total CDF for all containment types. The smaller contribution of ATWS sequences to the Limerick CDF is partly due to the several ATWS related enhancements carried out in Limerick (i.e., alternate rod insertion and automatic, two train standby liquid control), and may not be typical for other Mark II plants. The other sequences shown in Table 4.2, LOCA and TW, are less important and are negligible in the analyses of some plants. The SBO, ATWS, and transient sequences can be further divided into a fast sequence and a slow sequence. In NUREG-1150, a fast accident sequence is defined as one with core damage occurring in a short time after accident initiation (approximately 1 hour), and a slow accident sequence is defined as one with core damage occurring in the long term after accident initiation (approximately 12 hours).

Table 4.3 shows the timing of key events for some accident sequences. The values shown in the table are from calculations by the source term code package (STCP) [12, 14] or a combination of the BWR-LTAS, BWRSAR, and MELCOR [13] codes, and are typical for accident progression without any operator intervention. Table 4.3 shows that containment failure could occur at different times in different sequences. Although not shown in Table 4.3, the sequence of the key events for a TW sequence would be the same as that for the slow ATWS sequence, except that the time intervals between key events are much longer. Typical time to containment failure is expected to be greater than one day (30 hours for Peach Bottom), and typical time interval between vessel breach and containment failure is also very long (20 hours for Peach Bottom) because of the reduced decay power with time. The sequences of key events may be changed due to variation in plant parameters or by operator actions. For example, the containment would fail at vessel breach for the fast ATWS sequence presented in Table 4.3 if a containment failure pressure of 110 psig, instead of 130 psig, was assumed in the STCP calculation. The containment could also fail at, or before, vessel breach in a slow SBO sequence if the battery life was extended much longer (than six hours used in the calculation), or an alternate water supply was located and used for core injection.

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 4.3 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 II 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 [7].

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 (Table 4.4). The mechanisms that may cause significant SP boundary loads include (1) SRV air clearing and (2) SRV steam condensation. 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. SP temperature may also reach a level that may cause damage to pumps which take suction from the SP. This concern has been discussed in Section 3.1.2.2 and will be addressed later in strategies related to resource management.

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 temperature, and containment hydrogen and oxygen concentrations. 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 discussed in Section 3.2.1, 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.

SP Boundary Loads: The SP boundary loads are design assessment loads of a BWR containment, and, as such, the containment has been assessed for these loads under normal or design basis accident conditions, or operating procedures have been established to prevent plant conditions from reaching specific limits to ensure that unacceptable loading conditions would not occur, (e.g., the suppression pool level limits or the suppression pool heat capacity temperature limit, HCTL). However, during a severe accident, these limits may be exceeded due to the loss of certain plant safety functions. Furthermore, the loads that could occur concurrently with the SP boundary loads during a severe accident may be different from those used in the design assessment. For example, the containment may experience a high containment pressure load during this phase of a severe accident. A combination of the SP boundary loads and the high containment pressure load may result in containment failure earlier than expected. The strategies that can be used to eliminate these loads are presented in Table 4.4. More detailed discussions can be found in the Mark I report [7].

Containment Pressure Loads: The primary cause of an unacceptable containment pressure load is the lack of adequate containment heat removal (CHR) capability. This may occur either in an ATWS sequence or a TW sequence. The containment pressure rises vary rapidly in the ATWS sequence and as a consequence the containment may fail in a short time (Table 4.3). At the other end of the spectrum, the pressure rise in a TW sequence is much slower. The designed CHR capability is lost in a TW sequence either due to a failure of the RHR system to deliver water to the containment, or the loss of the RHR heat exchangers. Although containment spray is operational in the latter case, the delivery of the saturated suppression pool water to the containment atmosphere cannot repress containment pressure rise. However, containment sprays can be useful if an alternate cool water supply is used for the spray system. The spray of cool water into the containment atmosphere will slow down the containment pressure increase rate and thus extend the time available for recovery. (The use of an alternate water supply for the RHR system has been discussed in Section 3.2.1.) Additionally, the heat generated by the reactor core can also be removed from the reactor by in-vessel strategies such as the use of the reactor water cleanup (RWCU) system in its blowdown mode or the recovery of the main condenser as a heat sink. As a last resort, containment venting can be used to prevent, or delay, containment failure if the other strategies fail to prevent the containment pressure from rising. Important issues regarding containment venting have been discussed in Section 3.2.4.

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 presented in Table 4.5 and are discussed below.

4.2.2.1 Before Vessel Breach

The challenges to containment integrity during this time period include the SP boundary load and the containment pressure and temperature loads.

SP Boundary Loads: Suppression pool boundary loads 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, 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.

Containment Pressure Loads: The sources of containment pressure loads during this time period include the mass and energy released from the RPV to the containment and hydrogen combustion in the containment. They are discussed in the following.

Mass and Energy Addition During Core Degradation: The containment pressure load from mass and energy addition to the containment atmosphere is primarily due to the heat and gases generated in the RPV from decay heat and fuel cladding oxidation. This containment pressure load is not expected to cause a significant increase in containment failure probability. Analyses showed that containment failure pressure could be reached during this time phase in either a slow SBO sequence or an AE sequence (a large LOCA with loss of all core injection). In a slow SBO sequence, the combination of the pressure rise during core melt and the pressure rise before core melt (due to core steam generation from an extended battery operation) may exceed the containment pressure capability. The conditional probability of containment failure was estimated to be 1.6% in the NUREG-1150 analyses for Peach Bottom. Containment failure was avoided in some of the NUREG-1150 cases by containment venting, which was estimated to have a conditional probability of success of 11.6% [41]. In an AE sequence, containment overpressure failure was predicted to occur in Peach Bottom by STCP [42]. The release of the high temperature gases directly to the drywell (without the cooling by the suppression pool) results in a high containment pressure load, exceeding the containment pressure capability. This failure probability is not a significant concern, however, because of the small probability of the AE sequence and the uncertainties in load prediction [7].

In general, the containment pressure load will be smaller for a Mark II containment (than a Mark I containment) because of its larger volume. The failure probability for a Mark II containment is therefore expected to be smaller than that estimated for the Mark I containment (assuming the contianment strength is comparable). Although unlikely, there could still be a finite probability of containment overpressure failure during this time phase, and actions are required to mitigate this challenge. The containment cooling systems or the containment sprays (using an alternate cool water supply if necessary) can be used to remove the energy and steam in the containment atmosphere and thus reduce its pressure. However, noncondensible gases cannot be removed by these systems, and containment venting is required to maintain containment pressure if the use of the above systems is not sufficient to keep the pressure below the acceptable limit. Wetwell venting should be used because the containment atmosphere is significantly contaminated at this time.

Hydrogen Combustion: Hydrogen concentration in the containment atmosphere is influenced by the amount of fuel cladding oxidized during core degradation, which, in turn, depends on the core melt process. Without RPV depressurization, the amount of hydrogen generated during core degradation of a fast SBO sequence was predicted to be about 1,100 lb-moles (from the oxidation of approximately 53% of the fuel clad, 12% of the channel box walls, and 1% of the control blade stainless steel). Much less in-vessel hydrogen generation was predicted if the RPV was depressurized by ADS actuation according to the BWR EPGs (530 lb-moles from the oxidation of 23% of clad, 10% of channel box walls, and 1% of control blade stainless steel [13]). These predicted values (in lb-moles) are significant when compared with the amount of noncondensible gases initially in the containment (approximately 1,000 lb-moles for Limerick). With a large portion of this hydrogen released to the containment (about 50% for high RPV pressure cases and 100% for low RPV pressure cases) the containment hydrogen concentration would exceed its combustible limit, if sufficient oxygen is present. Since a Mark II containment is inerted, hydrogen combustion should not take place. However, oxygen may be introduced to the containment during an accident if instrument air, instead of instrument nitrogen, is used for equipment operation (Section 3.2.2). Oxygen can also enter the containment through the containment venting system (Section 3.2.1) or be generated by radiolysis. This can occur due to low containment pressure, as a result of containment venting and containment sprays (Sections 3.2.4 and 3.2.3).

In cases where the containment atmosphere's composition reaches the combustible limit, containment venting and purging can be used to alter the composition to below the combustible limit. For Limerick, the CAC system discussed in Section 3.2.1 can be used to supply nitrogen to the containment for containment venting and purging. However, the CAC system may not have sufficient pressure head and capacity to achieve an effective containment hydrogen control under severe core degradation conditions. Containment venting, although unable to change the containment atmospheric composition, can be used to reduce the amount of combustible gases in the containment and the containment pressure before hydrogen combustion. The use of containment venting can thus reduce the impact of hydrogen combustion by reducing the base pressure before hydrogen combustion and the amount of hydrogen burned. On the down side, hydrogen combustion during containment venting will result in a significant driving force for the release of the containment atmosphere and the fission products it contains, besides causing structural damage and igniting additional fires.

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 [43]. However, there is considerable uncertainty in droplet size in the containment during severe accidents. It is not clear whether droplets of size 20 microns or less would form in a containment atmosphere under conditions psotulated for a severe accident scenario. 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 combustion. According to the BWR EPGs, containment sprays are used when containment (either the drywell or the wetwell) hydrogen concentration reaches 6% and containment oxygen concentration is above 5%. The use of containment sprays as a means to mitigate hydrogen combustion is in general desirable but careful monitor and control of containment conditions is required to avoid any of the potential adverse effects discussed above.

Containment Temperature Load: High drywell temperatures are more probable during this time phase if the hot gases generated in the RPV are released directly to the drywell, bypassing the suppression pool. A very high drywell temperature was predicted for the AE sequence of Peach Bottom [42]. High containment temperature can fail the containment thermally or fail the containment by a combination of high pressure and high temperature (Section 3.1.1.1). Containment cooling and drywell spray can be used to cool the drywell atmosphere and reduce the probability of containment failure.

4.2.2.2 After Vessel Breach

As shown in Table 4.5, the loads that challenge containment integrity after vessel breach include (1) SP hydrodynamic loads, (2) containment pressure loads, and (3) transient pressure differential loads on the RPV pedestal created during high pressure melt ejection (HPME). These loads 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.

SP Hydrodynamic Loads: Immediately following a high pressure vessel breach, the pressure and temperature of the drywell atmosphere will increase rapidly. This pressure increase will clear the water column initially in the downcomer vent, cause a gas bubble to form at the exit of the downcomer vents, and result in a pool swell in the bulk mode (i.e., a slug of solid water accelerated upward by the air bubble). The mass and energy additions associated with the blowdown of the primary system after vessel breach are different from those of a 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.

Containment Pressure Loads: The mechanisms that can cause rapid containment pressurization and possible containment failure at vessel breach are (1) mass and energy addition to the containment atmosphere at vessel breach, (2) direct containment heating (DCH), and (3) fuel-coolant interaction (FCI). Early wetwell venting to reduce the initial containment pressure before vessel breach is suggested in Table 4.5 as a strategy to reduce the probability of containment failure from rapid containment pressurization for all of the above mechanisms. Since early venting may result in unnecessary fission product release if containment integrity can be maintained without venting, it should be used with extreme caution. Since VB is difficult to pinpoint, early venting could result in an open vent path at VB. This and other important issues on containment venting have been discussed in Section 3.2.4.

Mass and Energy Addition at VB: The degree of containment pressure rise resulting from a high pressure RPV blowdown depends on the amount of steam and hydrogen stored in the RPV and their temperature immediately before vessel breach. Containment failure at vessel breach may occur for an SBO sequence or a fast ATWS sequence due to the combined effect of high containment pressure before vessel breach and the containment pressure rise from the high pressure blowdown (about 50-70 psi, predicted by the STCP and MELCOR for fast and slow SBOs and ATWS [11, 13]). Although containment failure was not predicted to occur in either of the above sequences in Table 4.3, containment failure cannot be totally ruled out because there is considerable uncertainty regarding the containment pressure capability, the pressure immediately before vessel breach, as well as the pressure rise at vessel breach.

Drywell sprays, if activated before vessel breach, will condense steam and remove heat from the containment atmosphere and thus reduce the pressure rise during vessel blowdown. Early containment venting, before the vessel is breached, will reduce the initial containment pressure and, as a consequence, reduce the containment pressure load at vessel breach. Since this pressure load is caused by a high pressure vessel blowdown, it can be mitigated by maintaining the RPV at low pressure before vessel breach.

Direct Containment Heating (DCH): The pressure rise attributed to the above high pressure blowdown event may be augmented by the energy addition to the containment atmosphere from DCH. DCH refers to a series of physio-chemical processes that contribute significantly to the energy and mass input to the containment and thus the pressure rise in the containment atmosphere. 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' 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 whether there is sufficient time for the downcomer vents to clear. Since a large amount of aerosols, including refractory fission products, could be generated in high pressure melt ejection, significant release of radioactive material could result should the containment fail due to the DCH loading.

The mitigating effect of water in the reactor cavity for DCH (for plants with a deep reactor cavity, see Section 3.1.1.3) 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. However, 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 [4].

As in the mass and energy addition case, drywell spray can be used to reduce the containment pressure rise by condensing steam and removing energy from the containment atmosphere. The impact of drywell spray on dispersed core debris is not well understood and its effect on DCH is probably similar to that of the co-dispersed water discussed above. In general, the operation of drywell spray before vessel breach is believed to be beneficial and desirable. Early spray will ensure a substantial inventory of water in the reactor cavity , which would help mitigate the effect of DCH and promote the quenching of core debris. Additional strategies to mitigate the effect of DCH include early wetwell venting, which reduces the initial pressure in the containment and thus the impact of DCH, and RPV depressurization, which prevents HPME and can eliminate DCH.

Fuel Coolant Interaction (FCI): Two types of containment challenges may occur when the molten core debris contacts water: a steam explosion and a rapid steam generation, which of these is more likely to occur remains unceratin at the present time. In steam explosion, the rapid energy transfer from the fuel to the coolant results in an eruptive steam formation and a shock wave in the fuel coolant mixture. The containment loading conditions associated with a steam explosion thus include those from the vapor pressure, the shock wave, and the missiles generated by the explosion. In the other challenge, a rapid steam generation, the fuel coolant interaction does not exhibit shock wave characteristics. A large amount of steam is produced in rapid steam generation but the process is not explosive. The containment loading condition associated with this non-explosive steam generation is a quasi-static pressure load in the containment atmosphere [44].

The impact of FCI on a Mark II containment depends on the design of the region inside the reactor pedestal (Section 3.1.1.3). For those Mark II containments that have downcomers inside the pedestal (Shoreham and

NMP2), the core debris will flow down the downcomers and into the suppression pool. Significant FCI occurs when the core debris contacts the suppression pool water. The FCI loads that may cause containment failure include a dynamic pressure load (through the suppression pool) on the suppression pool boundary and a quasi-static pressure load on the containment walls. Both of these loads may fail the containment directly, or fail the reactor pedestal first and fail the containment indirectly. These loads could also fail the downcomers and create a suppression pool bypass condition. There are significant uncertainties on the nature and amplitude of the FCI loads, and analytical tools that can accurately predict FCI are still lacking.

For the Mark II plants that do not have downcomers in the pedestal region, the impact of the FCI loads depends on the geometry of the reactor cavity. They can in general be separated into two categories: those with a shallow reactor cavity and those with a deep reactor cavity. For the plants that have a shallow reactor cavity (Limerick and Susquehanna) the amount of water in the reactor cavity is limited, and the loads resulting from the interaction of the core debris and the water in the reactor cavity is not expected to be very severe. However, in addition to the above FCI that occurs in the drywell, FCI may also occur in the suppression pool for these plants. The core debris discharged from the RPV may overflow the reactor cavity, spread to the ex-pedestal region of the drywell, and consequently enter the suppression pool through the ex-pedestal downcomers. The subsequent FCI in the suppression pool will be similar to what happens in the plants with in-pedestal downcomers, but the loads are expected to be smaller because of the smaller amount of core debris involved in the FCI. The drywell FCI is most significant for the plants that have a deep reactor cavity (WNP2 and La Salle), if the reactor cavity is flooded at the time of vessel breach. A significant drywell FCI may fail the reactor pedestal and thus indirectly fail the containment, or fail the containment directly by a quasi-static pressure load. A wetwell FCI is not likely to occur during this phase of the accident for these plants.

Although there are substantial uncertainties in the determination of the FCI loads, significant loads could occur in most cases. Even if these loads may not fail the containment by themselves, a combination of these loads and the loads that already exists in the containment from other sources may result in containment failure. Strategies to prevent the occurrence, or mitigate the consequences, of the FCI loads are therefore needed. There does not seem to be any effective strategy, besides hardware modifications, for the wetwell FCI. On the other hand, the drywell FCI can be avoided by removing the water in the reactor cavity and preventing the use of any system that can add water to the reactor cavity, e.g., drywell spray. However, these actions are in direct conflict with other strategies that require the flooding of the reactor cavity or the use of the drywell spray. Early wetwell venting, as used in the other cases involving rapid containment pressurization, can be used here to reduce the initial containment pressure and thus the impact of pressure rise from FCI.

Transient Pressure Load on RPV Pedestal: The volume beneath the RPV (reactor cavity) is small and restricted (Figure 3.1). The primary communication passageway between this volume and the containment is one or two walkways. These have a limited area and will restrict the dispersal of the mass and energy from a high pressure blowdown. Consequently, there will be a transient pressure differential between the inside (reactor cavity) and the outside (drywell) of the reactor pedestal during vessel blowdown. The magnitude of this pressure load depends on the area of the walkways and the mass and energy input rate to the cavity. In general, the area of the walkways should be sufficiently large to keep the pressure differential low enough to prevent an excessive loading of the RPV pedestal. However, the time scale of vessel blowdown can be very short, and the mass and energy addition very large and thus cause a problem. In the NUREG-1150 analyses of Peach Bottom, a finite conditional probability for early containment failure was predicted to be caused by this load [41]. Since this load is caused by a high pressure RPV blowdown it can be prevented by RPV depressurization.

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 (if the accident is not a LOCA), 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 4.3, containment failure is predicted to occur in the late phase for both the slow and the fast SBO sequence and the fast ATWS sequence. The energy transfer from the hot core debris, and the hot gases released from the core-concrete interaction, cause the temperature and the pressure in the containment atmosphere to rise. The containment pressure may reach its failure pressure if mitigating actions are not taken in time. The drywell temperature is also very significant. It could reach about 1,000°F a few hours after vessel breach if containment cooling systems are not activated and CCI continues [13, 14]. Wetwell temperature will stay much lower than drywell temperature during CCI if the suppression pool is not bypassed. However, the probability of a suppression pool bypass for a Mark II containment could be very high [12].

CCI is the most important mechanism for containment loading in the late phase. As the high temperature core debris falls into the reactor cavity, 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 molten corium and generate a significant amount of noncondensible and combustible gases and release the chemical heat of reaction. The release of the high temperature gases and the transfer of heat from the hot corium to the containment atmosphere will result in significant pressure and temperature loads on the containment, and the release of combustible gases increases the probability of a combustion event. 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 amount and geometric configuration of the core debris on the floor, and the composition and material properties of the structural concrete.

In the Mark II containments, the design of the region inside the reactor pedestal has an important effect on the amount and geometric configuration of the corium on the drywell floor (Section 3.1.1.3), and consequently, a significant effect on the progress of CCI. Again, the Mark II plants can be separated into three categories according to the designs of the in-pedestal region. The plants that have downcomers in the pedestal region may have a smaller effect from CCI if most of the core debris remains liquid. In this case, the debris is directed to the suppression pool. The reactor cavity in the plants that have a deep cavity, is large enough to contain all the core debris discharged from the vessel. The confined geometry of the ex-vessel core debris will make the cooling of the core debris difficult. These plants may be vulnerable to pedestal floor failure. The plants that have a shallow cavity and no downcomers are likely to experience still another CCI scenario. The ex-vessel core debris in these plants may spread to outside the reactor pedestal, and some of the core debris may even flow to the suppression pool through the downcomers closest to the pedestal. The smaller amount of core debris remaining on the drywell floor, as compared with those plants having a deep reactor cavity, will result in shallower penetration. The greater contact area between the corium and the concrete floor due to corium spreading will cause a faster containment pressurization, but also less drywell floor penetration [14].

In addition to containment pressure and temperature loads, CCI may also create a suppression pool bypass condition for a Mark II plant. In the plants where corium can reach the downcomers, the corium may fail the downcomers upon contact, and the failure of the downcomers will create a bypass condition. In the plants where there is a deep reactor cavity and the corium can not reach the downcomers, suppression pool bypass is still likely, because all these plants (in fact, all Mark II plants with the exception of Susquehanna) have drains (into the wetwell) inside the reactor cavity, and these drains could fail by corium attack a short time after vessel breach (about 20 minutes) [12]. A suppression pool bypass will increase the containment pressure load due to the loss of suppression cooling and condensation. It will also have a significant effect on containment venting. If there is a suppression pool bypass containment venting is likely to be less desirable at this time, when the radioactivity in the containment atmosphere is high, and the SP fission product scrubbing capability is lost.

Although the effect of CCI is reduced if some of the corium is transferred to the suppression pool, the containment pressurization rate may not be reduced. The corium entering the suppression pool will generate steam and cause containment pressurization. Without even considering a steam explosion, MELCOR calculations showed that the containment pressure could increase at a faster rate with more core debris entering the suppression pool. (Because of modelling limitations, a fraction of the core debris leaving the failed reactor vessel was deposited directly into the wetwell water pool in the MELCOR calculations [13].) For a fast SBO sequence with ADS actuation, MELCOR predicts a containment overpressure failure at about 8.5 hours after accident initiation if all corium is assumed to enter the suppression pool. Conversely, the containment pressure capability is not reached 13.5 hours after accident initiation (when the drywell floor is penetrated) if all corium is assumed to remain in the cavity. Results from MELCOR calculations also indicate that the time to reach a high containment pressure (100-120 psig) is maximized for some intermediate debris split (between drywell floor and suppression pool) [13]. It is noted that, even though the steam generated in the suppression pool may contribute significantly to the containment pressure load, the steam may have a moderating effect on the containment temperature load, because the temperature of the steam is much less than the temperature of the gases released from CCI. In addition, condensation heat transfer between steam and containment structures is more effective than convection heat transfer between non-condensibles and structures. Furthermore, the fission products released from the submerged corium to the containment atmosphere are also significantly reduced because of the scrubbing of the suppression pool.

As mentioned above, the drywell floor may be penetrated before the containment pressure capability is reached. A penetrated drywell floor will create a suppression pool bypass and result in a FCI when the corium falls into the wetwell. (This will be the case for all but one Mark II plant. In the La Salle Plant where the pedestal is solid concrete up to about the normal water level in the wetwell region, CCI will continue in the wetwell after the drywell floor is penetrated.) The time to drywell floor penetration depends on drywell floor thicknesses, which for Mark II plants vary from three to five feet, and other parameter values (e.g., concrete composition) and the computer codes used in the calculation. Whether drywell floor penetration occurs before or after containment failure is uncertain. For a sequence where all core injections are lost at the beginning of the accident and the RPV depressurization is successful (a TQUV sequence), the STCP predicts a containment overpressure failure at about 8.5 hours after accident initiation, before the drywell floor is penetrated [17]. On the other hand, for a similar sequence (a fast SBO with RPV depressurization), MELCOR predicts drywell floor penetration before containment pressure failure (The containment pressure at drywell floor penetration, occurring at 13.5 hours after accident initiation, is about 120 psia) [13]. The attack, and eventual penetration, of the drywell floor by the corium may result in several loading conditions: the weakening of the reactor pedestal may cause a gross motion of the reactor vessel and consequential containment failure; the interaction between the corium and the suppression pool water may result in a steam explosion and failure of the reactor pedestal or the containment; and the steam generated in the suppression pool will cause continuous containment pressurization and possible containment overpressure failure.

The potential challenges to containment integrity and their mechanisms during this phase of an accident as discussed above are summarized in Table 4.6, along with the strategies for mitigation. More detailed discussions on these items can also be found in the Mark I report [7].

Containment Pressure Loads: The sources of containment pressure loads include the heat and noncondensible gases generated during CCI, or the steam generated from FCI after the corium is relocated to the suppression pool. Containment pressure load may also arise from the combustion of gases. The challenge of gas combustion and the corresponding mechanisms and strategies are the same as those discussed under hydrogen combustion in Section 4.2.2.1 and will not be repeated here. The strategies to mitigate the containment pressurization due to other sources are discussed in the following.

Containment cooling is one of the strategies that can be used to mitigate the containment pressure load due to mass and energy addition (Table 4.6). It 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. Its effect on the containment pressure load during CCI has been discussed in the Mark I report using results from STCP calculations with a drywell spray at various flow rates. Drywell temperature is significantly reduced by the spray due to its energy removal capability, and drywell pressure is decreased accordingly.

Containment cooling cannot prevent a further 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 wetwell venting should be used unless the suppression pool is bypassed. It should be noted, however, that containment venting during CCI, when high temperature gases are released to the containment atmosphere, may result in higher containment temperature, and thus earlier containment failure (than the no venting case) by combined temperature and pressure loads[1] [7]. Additional energy removal from the containment atmosphere is needed to prevent containment failure in this situation. Such removal can be achieved by the operation of the drywell spray.

Since CCI is a major source of the containment pressure load, the control of its progression is an important consideration. 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 in a coolable configuration. Water may be introduced to the reactor cavity before vessel breach by the drywell spray. However, as discussed above, water in the reactor cavity increases the probability of an ex-vessel steam explosion (Section 4.2.2.2).

A continuous supply of water to the reactor cavity can keep the core debris flooded and cooled. The water that is added to the RPV after vessel breach will flow through the break to the reactor cavity. Water can also be added to the corium through the drywell (e.g., by the drywell spray). This is more easily achieved, and the effect more significant, for the plants that have a shallow reactor cavity, where corium is likely to spread to the drywell region outside the reactor pedestal. A drywell water supply not only reduces the containment pressure load by reducing the progression of CCI, but also has the potential to prevent a suppression pool bypass by preventing downcomer failure due to corium attack. Water addition through the drywell is more unlikely for the plants that have a deep cavity. Although there may be communication paths between the ex-pedestal drywell floor and the in-pedestal cavity, these paths may be plugged during CCI and adding water to the corium through the drywell then becomes impossible. Even if water can be delivered to the corium, the effect of water on the progression of CCI may not be significant because the corium layer in the reactor cavity is thick and confined. However, water addition is still desirable because the overlying water can cool the gases released from CCI and scrub the fission products. On the downside, the addition of water to the corium on a dry floor may cause an initial pressure increase due to steam generation. However, the pressure increase is in general expected to be moderate and the benefit from water addition is expected to be more important.

Continuously adding water to the containment using water sources external to the containment will result in a containment flooding. Containment flooding has been suggested as an in-vessel strategy to cool core material in the vessel by removing heat through the bottom head if injection cooling proves unsuccessful. From the ex-vessel point of view, once flooded, cooling can be provided to the corium with minimal use of active equipment, e.g., pumps.

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[1]The results presented in Reference 7 were obtained from STCP calculations. Since the drywell was modelled as a single volume in STCP, the effect of drywell source (or sink) locations and thermal stratification is not accounted for in the analyses. A multiple volume code, which can account for buoyancy, e.g., MELCOR, is required to provide a more detailed drywell temperature load profile. The drywell temperature loads will depend on the source (e.g., gases released from CCI) and sink (e.g., downcomer vent entrance if the wetwell is vented) locations and the corresponding circulation currents established in the drywell atmosphere. The consequence of the drywell temperature loads will depend on the temperature stratification and the relative location of containment areas that are vulnerable to high temperature load (e.g., seal area).

Strategy Identification

Containment flooding is also a part of the RPV control guideline in the BWR EPGs (Contingency #6)[2] [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 (up to the level of the top of active fuel, TAF), 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.

As water is added to the suppression pool, the gases in the wetwell airspace will be displaced to the drywell through the vacuum breakers between the drywell and the wetwell until the drywell-wetwell vacuum breakers are flooded. A portion of the wetwell atmosphere will be trapped at the top of the wetwell, and as a consequence, the water required to flood the wetwell is less than the total wetwell airspace volume (155,400 ft^3 for Limerick, Table 3.2). Additional water is required to flood the drywell to cover the corium, or to the top of reactor core if substantial core material is still in the RPV.

The time required for containment flooding depends on the level of flooding required and the water supply systems available at the time of the accident. For Limerick, there is no special system for suppression pool makeup, and the systems that provide normal and alternate water supplies, as discussed in Section 3.2.1, are used for containment flooding. Assuming one train of the RHRSW system is available for supplying water to the SP (normally one of two 100% capacity pumps, or 9,000 gpm), the time required to flood the containment up to the drywell floor (approximately 155,400 ft^3, depending on the vacuum breaker location) would be approximately two hours. Another two hours are required to flood the drywell up to the bottom of the RPV (approximately another 135,000 ft^3). The time will be shortened if larger, or additional water supplies are available for containment flooding. However, the water available for containment flooding may be limited by the capacity of the water sources, although additional water sources can be made available as discussed in Section 3.2.1.

During containment flooding, the free volume of the containment is significantly reduced and the containment atmosphere is correspondingly compressed. The displacement of 155,400 ft^3 free air space by water is more than one third of the original air space volume of the whole containment (The combined drywell and wetwell air space is about 404,100 ft^3 for Limerick, Table 3.2.). Even if the containment is flooded only up to the level of covering the corium on the drywell floor, containment free volume will be reduced to approximately two thirds to one half its original value, and the pressure will correspondingly be increased by a factor of 1.5 to 2. Wetwell venting can be used to maintain or reduce containment pressure before wetwell vent paths are flooded, and drywell venting, preferable with the use of the SGTS, can be used after the wetwell vent paths are flooded.

During containment flooding, there is a time interval, after the wetwell vent paths are flooded but before the corium on the drywell floor is completely covered with water, in which the containment atmosphere if vented, will not have the benefit of fission product scrubbing by a water pool. Preplanning should ensure that drywell venting is not needed in this time interval, or that the drywell spray or the SGTS can be used for fission product removal if drywell venting during this time interval cannot be avoided.

Mass and energy will be continuously added to the containment air space from decay heat and CCI, if CCI is not terminated by the flooding, and containment pressure will continue to rise. It is important that the drywell vent paths are not flooded as they are needed for containment pressure control. Since the core materials are flooded, the release through the drywell vents will have been subjected to pool scrubbing. The SGTS can be used to further reduce the release of the radioactive material to the environment.

[2]The objective of the BWR EPG Contingency #6 is to keep the reactor core covered through a break in the primary system pressure boundary. This is in contrast to the proposed in-vessel strategy mentioned above, where cooling to the reactor core is provided by heat transfer through the vessel. Contingency #6 is required when RPV water level cannot be restored and maintained above TAF (from Contingency #1), when the minimum alternate RPV flooding pressure cannot be established (from Contingency #4), or when RPV Water level cannot be restored and maintained above the steam cooling RPV water level (from Contingency #5).

Drywell Temperature Load: Drywell temperature can be significantly higher than the design temperature if mitigating actions are not taken in time. As a consequence, the containment may fail due to the high temperature or a combination of temperature and pressure loading (Section 3.1.1.1). Drywell temperature can be controlled by removing energy from the drywell atmosphere or by quenching the corium. Some of the strategies discussed above for containment pressure control can be used for drywell temperature control as well, e.g., containment cooling and drywell spray can be used to remove containment energy, and flooding the containment to keep the corium submerged can moderate and eventually terminate CCI.

Drywell Floor Melt-through or RPV Pedestal Erosion: As discussed above, drywell floor melt-through or RPV pedestal erosion could challenge the containment integrity directly or indirectly. Since these challenges result from concrete attack by hot corium, they can be moderated by quenching the corium. Corium and containment flooding discussed above can be used to achieve this purpose.

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, high temperature, or high water level in the secondary containment. Both the emergency response facilities and the radiological emergency response plans (Section 3.3) would probably have been activated before the accident reaches this phase. The secondary containment and radioactivity release control guidelines of the EPGs (Sections 3.3.2.2. and 3.3.2.3) would also have been previously 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 FP release after containment failure.

The challenges, mechanisms, and strategies during the release phase are shown in Table 4.7. For some of the challenges, the conditions that could lead to a FP release exist before an actual release occurs and actions to mitigate these challenging conditions can be taken either before or after containment failure (CF). For example, reducing the amount of fission products in the containment atmosphere before CF will reduce the potential for FP release should CF actually occur and is thus desirable. Reducing the containment pressure before CF is also desirable because it reduces the driving force for FP release should the containment fail later. For other challenges, diagnostic and mitigating actions are possible only after FP release has started. A detailed discussion of these challenges and strategies is presented in the following.

FP Concentrations in the Containment Atmosphere: 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 other less volatile FP groups will be a small fraction of their inventory.

For Limerick and other Mark II plants the fission products from the in-vessel release will pass through the suppression pool (by SRV actuation) as long as RCS integrity is maintained. As a result, a significant fraction of the non-noble gas fission products will be retained in the suppression pool (suppression pool decontamination) and the impact on environmental consequences may not be important (Section 3.1.2.1). The in-vessel release will be more important for those sequences where RPV inventories are discharged directly to the DW, e.g., sequences with a stuck-open vacuum breaker on the SRV tailpipe or LOCA sequences.

It is desirable to have the in-vessel release passing through the suppression pool, preferably through the SRV spargers, before discharging to the containment airspace. 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 delaying valve closure after each actuation (thus reducing the number of actuation required). For sequences where fission products are released directly to the drywell, without passing through the SP, drywell spray can be used to remove fission products from the drywell atmosphere (Section 3.2.3).

At vessel breach, the fission products in the RPV will be released directly to the drywell. The FP release is most significant if the primary system is pressurized at the time when the bottom head of the RPV is breached. The molten core debris may be ejected under pressure and result in significant aerosol generation and fuel fragmentation. The fission products discharged to the drywell atmosphere include those originally in the RPV atmosphere inventory and those released from the core debris during HPME. RPV depressurization before vessel breach will result in both SP scrubbing for the fission products originally suspended in the RPV and the elimination of aerosol generation associated with HPME. An ex-vessel steam explosion (EVSE), if it occurs, may also cause a significant FP release to the containment. EVSE can be avoided by eliminating water in the reactor cavity as discussed above. Drywell spray, if actuated before VB, can reduce containment airborne fission products by scrubbing.

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 structure 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 a large amount of 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. The drywell spray can add cool water to the drywell floor and thus reduce the release of iodine from the water pool covering the drywell floor. Containment flooding would provide a large volume of water and a corresponding reduction of iodine concentration and release.

Natural deposition will remove airborne fission products from the containment atmosphere. In absence of other sources, 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.

<u>Driving Force for FP Release</u>: Containment pressure provides the driving force for FP release after CF. The containment atmosphere and the fission products it contains will be released directly to the outside of the

containment, without the benefit of SP scrubbing, if the containment fails in the drywell. Early wetwell venting, which has been suggested in the previous sections as a strategy to reduce the probability of CF, can also be used to reduce the driving force for FP release in cases where significant DW failure potential exists or even after CF. Wetwell venting will provide SP scrubbing and reduced FP releases if subsequent drywell failure is inevitable.

FP Release From Containment to Outside Containment: 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.

In certain modes of containment leakage, the magnitude of the leak area increases with containment pressure (NUREG-1037, [17]). The total pressure-dependent leak area for Limerick, including that from the drywell head, equipment hatches, and suppression chamber access hatches, is estimated in NUREG-1037 to be less than 0.005 in^2 at 55 psig, 1.3 in^2 at 85 psig (1 in^2 in the drywell and 0.3 in^2 in the wetwell), and 42.9 in^2 at 140 psig (42 in^2 drywell and 0.9 in^2 wetwell). Among these potential leak areas leakage at the drywell head is one of the most important early CF modes identified in NUREG-1150 [4]. Such a failure will result in a direct leak path from the containment to the upper part of the secondary containment (SC), bypassing the suppression pool and most of the secondary containment.

Since the volumetric flow rate from an 1 in^2 leak area is approximately one containment volume per day for a Mark II containment, containment depressurization from the above leak areas could be slow (in terms of hours). To reduce the total amount of release from the drywell leak areas, wetwell venting can be used to accelerate the pressure reduction and possibly to reclose the drywell leak areas. Although the total amount of containment atmosphere released to the outside is not reduced by wetwell venting, FP amounts released to the environment are reduced. Additional release reduction may be achieved by selecting a wetwell vent path that provides a greater secondary containment retention capability and a more favorable release point.

The FP release will also be reduced if the leak area can be flooded. The flow from the containment atmosphere will then pass through a pool of water where some of the fission products will be retained. SC radiation and temperature monitoring systems can be used to identify the leak location. Analytical results on containment performance such as those presented in NUREG-1037 can also provide information about possible leak sites and ways to flood these areas. One particular leak area that warrants special attention is the drywell head. Flooding of this location in the reactor refueling well area, using the water source and systems designed for reactor refueling, will have a significant beneficial effect on FP release and offsite risk. Its feasibility for individual plants warrants further investigation.

A containment isolation failure may result in the leakage of radioactive material to the secondary containment or directly to the environment. The probability of this failure mode is low for Mark II containments because a failure of some parts of the containment isolation system can be identified during plant operation from excessive nitrogen requirements for containment inerting. However, if an isolation failure should occur, the BWR EPGs provide 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.

FP Release From the RCS to Outside the Containment: In a containment bypass sequence (V sequence), the radioactive material in the RCS can escape directly to the reactor building or the environment. This may occur 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 (an interfacing systems LOCA, ISL). It is a very unlikely, but high consequence, event. 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.

FP Release From the SC to the Environment: According to the BWR EPGs, the secondary containment (SC) HVAC should be isolated and the SGTS initiated if the secondary containment HVAC exhaust radiation level exceeds the isolation setpoint. The system will also be isolated upon receipt of a plant isolation signal. The SGTS then provides a filtered release of the secondary containment atmosphere to the environment at an elevated location. Direct release to the environment is prevented because the SGTS maintains the secondary containment at a negative 0.25 inch water gauge pressure.

The combination of SC isolation and SGTS operation can significantly reduce the release of radioactive materials to the environment if the flow from the primary containment (or the primary system if there is a bypass) 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 SC will increase, and leakage directly to the environment, as well as failure of the SC blow-out panels, or even structural damage to the SC or the SGTS, may result. Even under conditions where substantial leakage from the SC develops, the operation of the SGTS 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. The flow through the SGTS, and thus the benefit of filtering, could be enhanced if the SGTS can be operated in a recirculation mode.

The performance of the SGTS is also limited by the amount of aerosols collected in the high efficiency particulate air (HEPA) filters. The pressure drop across the filter increases as the amount of aerosols collected on the filter increases. The HEPA filter could be ruptured when the pressure drop across the filter exceeds its design limit.

The above limitations indicate that the effectiveness of the SGTS will be reduced if the rate of gases leaked or vented into the RB is large or if aerosol content in the gas discharge is high. In the containment performance improvement program, a hardened vent path bypassing the SGTS is suggested for containment venting. This is particularly helpful for containment venting early in an ATWS sequence when the required venting flow rate is large and the fission product content is low. The aerosol content in the containment atmosphere will be high if the aerosols generated from core degradation are released directly to the drywell, or if no mitigating actions are taken to remove the aerosols generated from CCI during the ex-vessel release. Because of the limited capacity of the HEPA filters for aerosol loads, aerosol scrubbing by the SP or containment sprays should be utilized to reduce the amount of aerosols released to the SC and thus the aerosol load on the SGTS filters.

It is important to maintain the SGTS functional throughout an accident. It has the capability to remove volatile forms of iodine, e.g., elemental iodine, from the release. Such volatile forms of iodine cannot be removed by pool scrubbing or deposition, and the potential for their release late in an accident from the water pools in the containment exists. The SGTS may lose its function or its effectiveness from structural damage. Possible sources of structural damage and methods to avoid such damage have been discussed above. In addition to structural damage, the SGTS may also lose its function if its fire dampers close.

If a significant amount of fission products is released to the secondary containment, the SC fire spray system, if manually operable, can be used to scrub fission products from the SC atmosphere and consequently reduce the release of fission products to the environment.

4.2.5 Fission Product Release During Containment Venting

The objective of the strategies described in Sections 4.2.1 and 4.2.2 is to prevent containment failure and the resultant uncontrolled fission product release. Containment venting has been suggested as a potential strategy to prevent containment failure during various accident situations. The implementation of containment venting is usually based on the assumption that containment failure is inevitable without venting and that the consequence of a controlled release would be less severe than that from containment failure. The decision to initiate venting thus involves projection of future accident progression. To make optimum accident management decisions, tools for this purpose should be readily available to the plant operation personnel.

CRETs discussed in this report could be used as such a tool for accident projection, the results of which can be used to make CRM decisions. The data for the important events used for accident projection during an accident would be different from those used for pre-accident PRA studies because some events have already occurred. Some plant conditions may be obtained from plant instruments and more appropriate probability distributions may be inferred for some of the crucial events.

For example, using probability data that is updated as the accident is in progress to quantify the probability of recovery of equipment and resources, and using the short term forecast of meteorological conditions, and the existing and expected offsite emergency response data, would result in a more reliable data base than that available from pre-accident PRA analyses, for making CRM decisions. Parametric studies using the above approach could provide valuable information for SAM decisions. Using this technique, containment venting decisions could be based on predictions of the expected consequences both with and without containment venting. Additional information may be needed to help the decision for venting. This may be derived from additional analyses addressing timing and pressure rise rate.

4.3 The Safety Objective Tree

The results of the above strategy identification effort are summarized in the safety objective tree shown in Figure 4.2. 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 4.2 was constructed according to the process defined in Section 2.2 and the results of strategy identification presented in Tables 4.4 through 4.7. It systematically defines the challenges to the overall safety objectives for a Mark II containment, identifies safety functions that need to be preserved to meet the objective and lists the specific challenges found in a Mark II 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 4.2, 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 Comparison of BWR Mark I, II, and III Containments

	Limerick (Mark II)	Peach Bottom (Mark I)	Grand Gulf (Mark III)
Reactor Type	BWR/4	BWR/4	BWR/6
Rated Power (MWt)	3,293	3,293	3,833
Drywell Free Volume (ft^3)	248,700	159,000	270,000
Wetwell Free Volume (ft^3)	155,400	127,000	1,264,000
Suppression Pool Water Volume (ft^3)	124,700	122,900	136,000
Containment Design Pressure (psig)	55	56	15
Drywell Design Temperature (°F)	340	281	330
Wetwell Design Temperature (°F)	220	281	185
Estimated Containment Pressure Capability (psig)	140	150*	55*

*50th percentile value in NUREG-1150.

Table 4.2 Important Plant Damage States and their Contribution in Percentage to the Total Plant Core Damage Frequency

Plant Damage State	Limerick[1]	Peach Bottom[2]	Grand Gulf[2]	Generic Mark II[3]
CDF (1/yr)	1.5E-5	4.5E-6	4.0E-6	1.3E-5
SBO	44.4	46.7	97.1	41.6
ATWS	7.7	43.1	2.9	27.9
Transient	37.3	5.0	--	27.7
LOCA	0.4	5.8	--	-
TW	5.0	--	--	2.7

Note:
(1) Based on the top 20 core damage sequences (95% of total CDF) of Limerick PRA.
(2) Data from NUREG-1150.
(3) Data from NUREG/CR-5528.

Table 4.3 Typical Timing for Different Accident Phases of Various Accident Sequences

	Very Early Phase		Early Phase		Late Phase
Accident Sequence	Accident Initiation (hr)		Onset of Core Melt (hr)	Vessel Breach (hr)	
Slow SBO(Note 2)	0		12	16	CF (21)(Note 1)
Fast SBO(Note 3)	0		2	4	CF (10)
Slow ATWS(Note 4)	0	CF (0.7)	1	2.5	
Fast ATWS(Note 5)	0		0.5-1	2	CF (3)

Notes:

1. CF (15) means containment failure occurs at 15 hours after accident initiation.
2. The plant batteries last 6 hours after accident initiation.
3. All core injection is lost at the inception of the accident and the ADS is not actuated.
4. It is assumed that the core power is at 30% normal power level and the containment failure causes the loss of core injection systems.
5. Core power is assumed at 30% normal power level. Core injection is lost before containment failure. Core melt may begin in less than 0.5 hour if core injection is lost at accident initiation.

Table 4.4 Challenges, Mechanisms, and Strategies During the Very Early Phase

Challenge	Mechanism	Strategy
1. SP Boundary Load	SRV Air Clearing with High SP Water Level	SP Water Level Control or RPV Depressurization or RPV Cooldown
	SRV Steam Condensation with SP Temperature Exceeding Pool Temperature Limit	SP Cooling or RPV Depressurization
	Downcomer Vent Chugging Load	Drywell Spray to Redistribute Noncondensible Gases
2. Containment Pressure Load	Loss of Vapor Suppression Capability	Restore SP Capability, Containment Spray, or RPV Depressurization
	Inadequate Containment Heat Removal Capability	Containment Cooling by Restoring CHR or by Containment Spray Using an Alternate Cool Water Supply, or Containment Venting

Table 4.5 Challenges, Mechanisms, and Strategies During the Early Phase

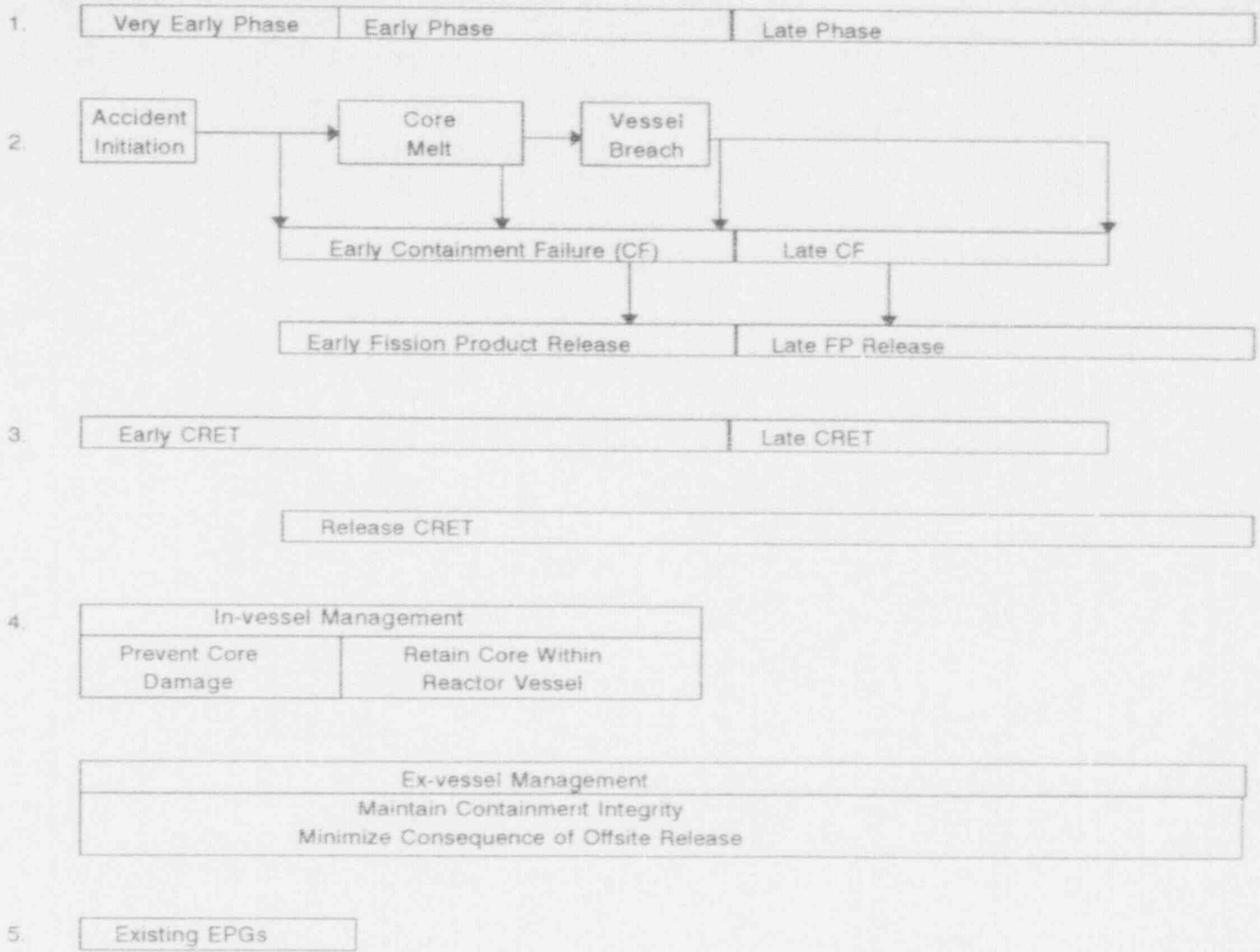
Challenge	Mechanism	Strategy
Before Vessel Breach		
1. SP Boundary Load	SRV Discharge with High RPV Noncondensible Gas Content	RPV Depressurization
2. Containment Pressure Load	Mass and Energy Addition (Noncondensible Gas Buildup)	Containment Cooling, Containment Spray, and Wetwell Venting
	Burning of Combustible Gases	Maintaining Containment Inerting, Wetwell Venting to Reduce Initial Pressure, Vent and Purge, or Containment Spray
3. Drywell Temperature Load	Mass and Energy Addition	Containment Cooling and Containment Sprays
At Vessel Breach		
1. SP Hydrodynamic Load	Pool Swell from RPV Blowdown with Primary System Conditions Different than that of LOCA	RPV Depressurization
2. Containment Pressure Load	Mass and Energy Addition at VB	Drywell Spray or RPV Depressurization
	Direct Containment Heating (DCH)	RPV Depressurization, Drywell Spray, or Flooding Reactor Cavity
	Ex-vessel Steam Explosion	Eliminate Water in Reactor Cavity
	(For Above Three Mechanisms)	Early Wetwell Venting to Reduce Initial Pressure
3. Pressure Load on RPV Pedestal	Rapid Mass & Energy Addition to the Inside of RPV Pedestal	RPV Depressurization

Table 4.6 Challenges, Mechanisms, and Strategies During the Late Phase

Challenge	Mechanism	Strategy
1. Containment Pressure Loads	Mass and Energy addition (CCI or FCI)	Containment Cooling, Wetwell Venting or Small Area Venting with SGTS, Corium and Cavity Flooding, Drywell Spray, or Containment Flooding with Drywell Venting
	Burning of Combustible Gases	Wetwell Venting to Reduce Initial Pressure, Vent and Purge, or Containment Spray
2. Drywell Temperature Load	High Temperature Gas and Energy Addition	Containment Cooling, Drywell Spray, Corium and Cavity Flooding, or Containment Flooding with Drywell Venting
3. Drywell Floor Meltthrough or Pedestal Erosion	Thermal Attack by Core Debris	Drywell Spray, Corium and Cavity Flooding, or Containment Flooding with Drywell Venting

Table 4.7 Challenges, Mechanisms, and Strategies During the Release Phase

Challenge	Mechanism	Strategy
Before or After CF		
1. FP in Containment Atmosphere	In-vessel Release (Core Melt)	SP Scrubbing or Drywell Spray
	Release During VB	RPV Depressurization or Drywell Spray
	Ex-vessel Release (CCI)	Corium and Cavity Flooding, Drywell Spray, or Containment Flooding
	FP Revolatilization from RCS	Adding Water to RPV, Containment Cooling, Drywell Spray, or Containment Flooding
	Late Release of Iodine from Water Pool	SP Cooling, Drywell Spray, Containment Flooding, or Adding Base to SP
2. Driving Force for FP Release	Containment Pressure	WW Venting or Small Area Venting with SGTS
After CF		
1. FP Release from Containment to Outside Containment	Containment Failure or Venting	WW Venting, Flooding Leak Area, or Containment Flooding
	Isolation Failure	Isolating Failed System, or Same as Above for CF
2. FP Release from RCS to Outside Containment	Interfacing Systems LOCA (ISL)	Isolating Break Area, RPV Depressurization, or Same as Above for CF
3. FP Release from Secondary Containment (SC) to Environment	Pressure and FP Increase in SC	SC Isolation, SC Fire Spray, or Using SGTS



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

Figure 4.1 Time Phases of Accident Pr‿gression and Accident Management

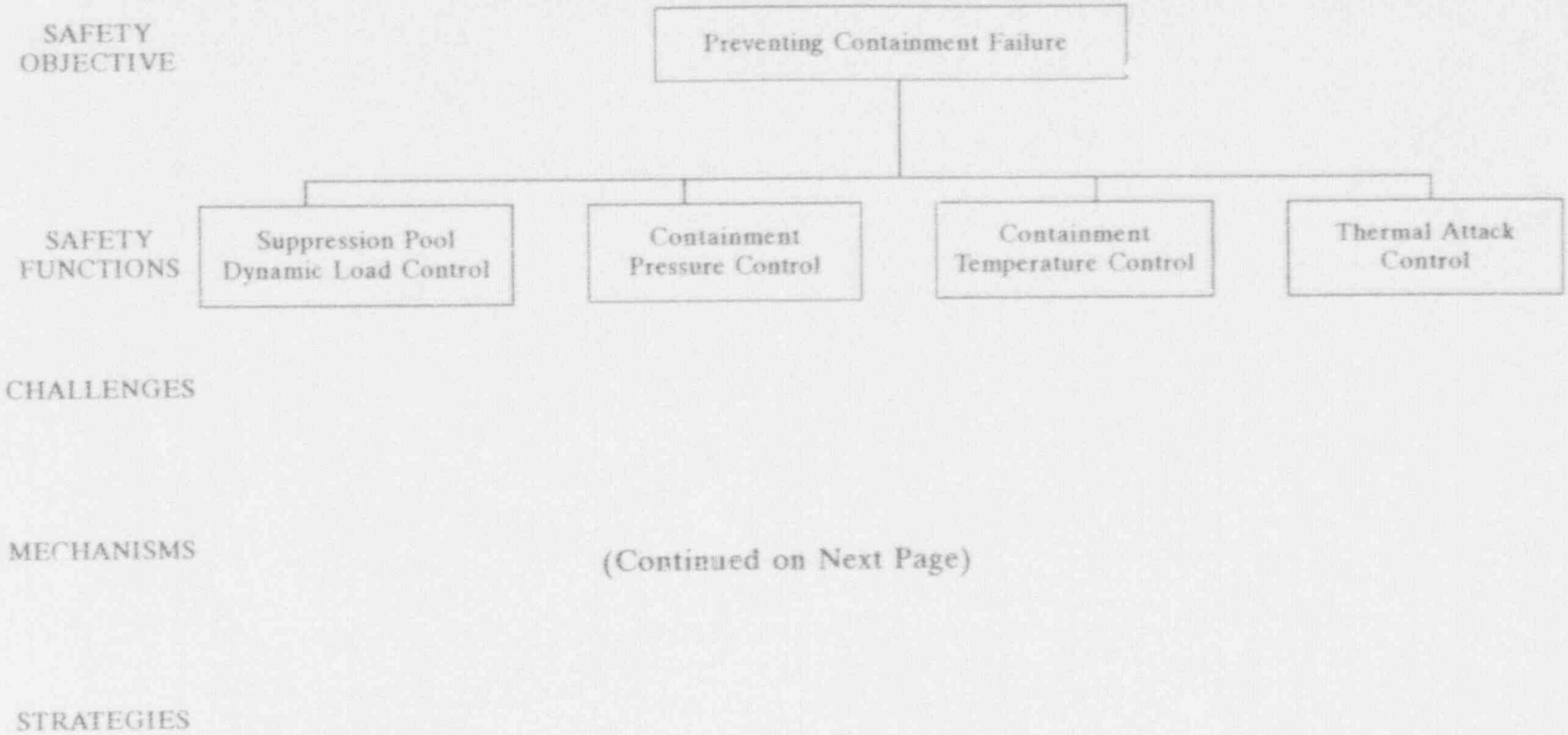


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Figure 4.2 Safety/Objective Tree with Identified Strategies for a Mark II Containment

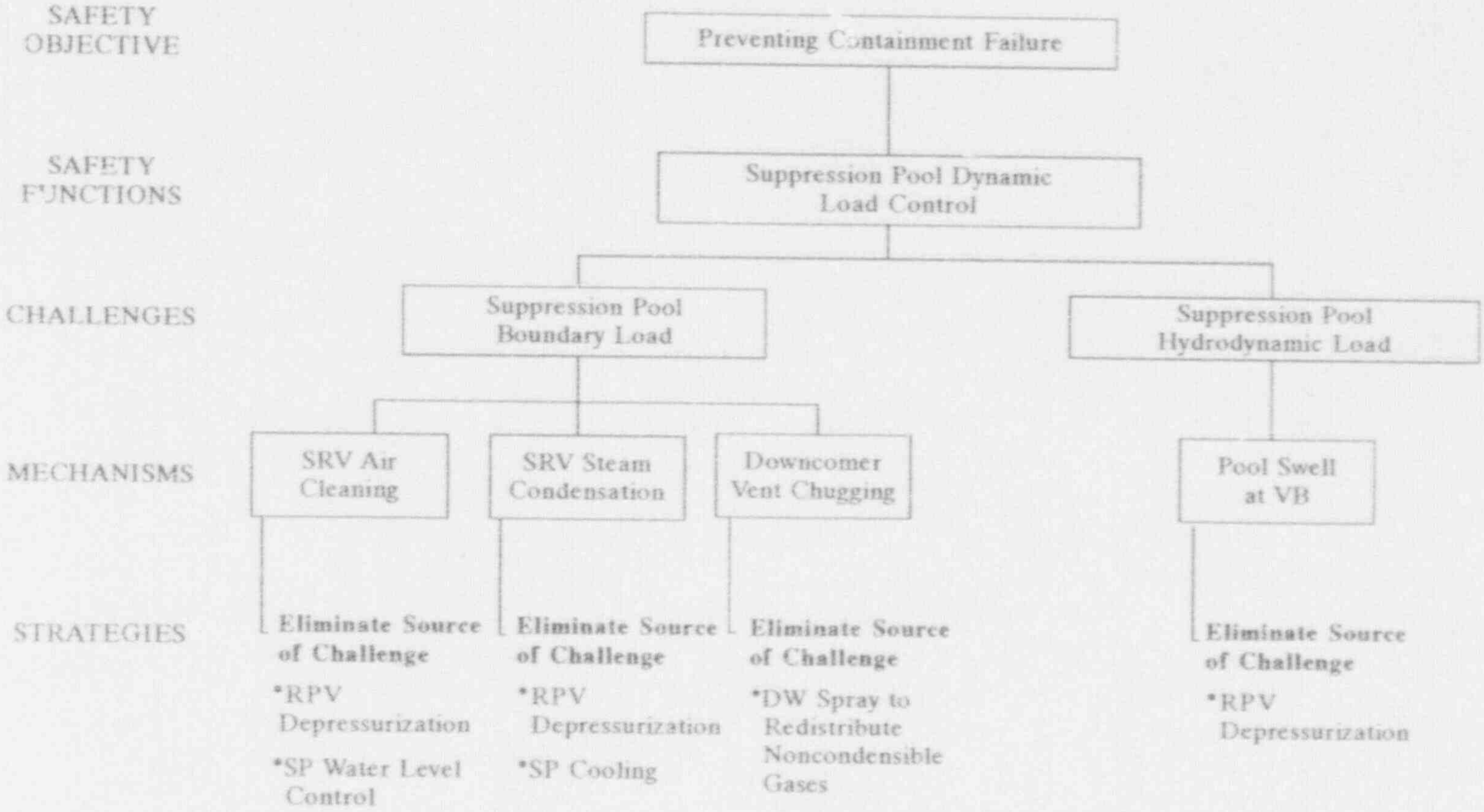

Figure 4.2 Safety/Objective Tree with Identified Strategies for a Mark II Containment (Continued)

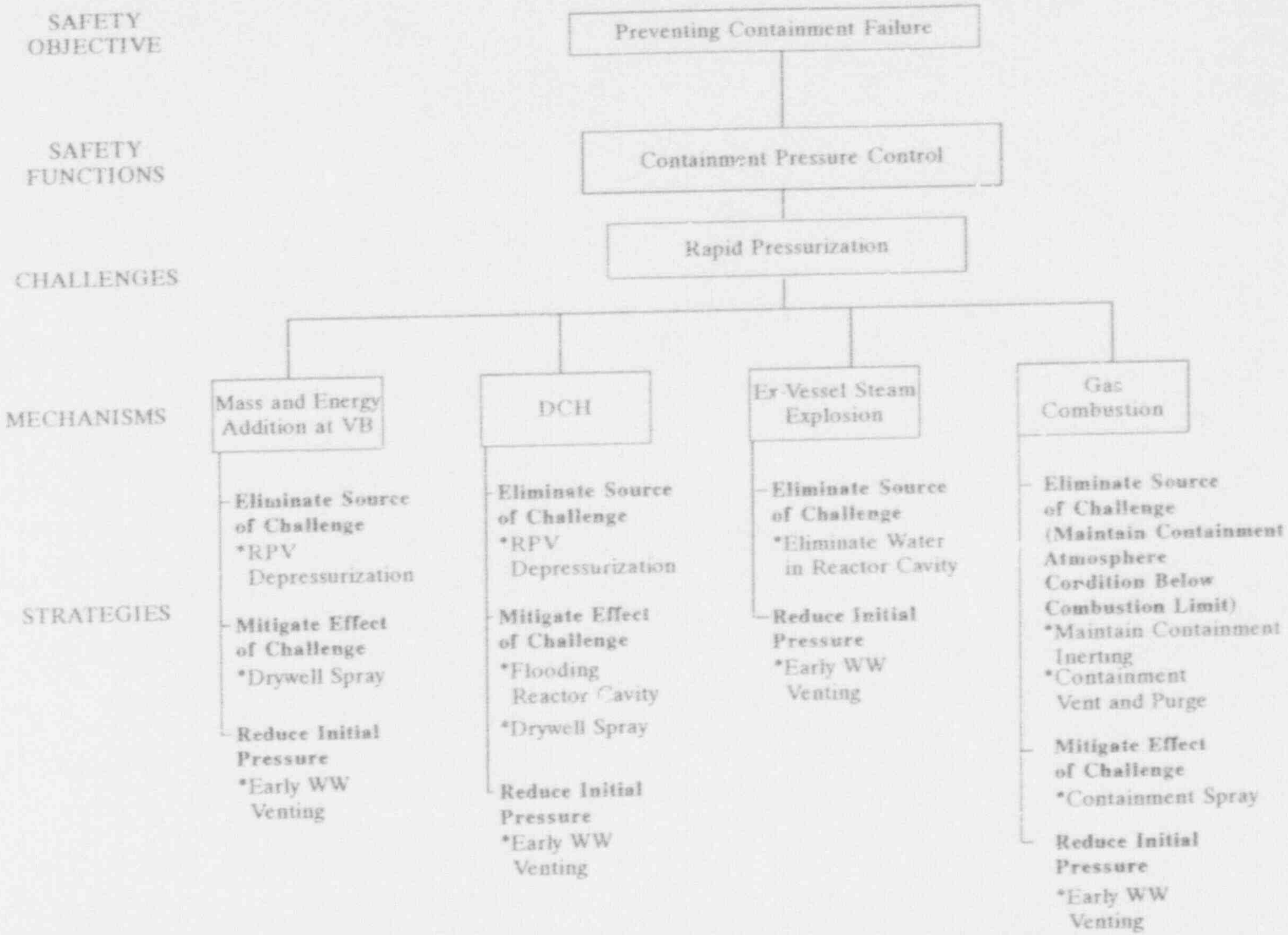

Figure 4.2 Safety/Objective Tree with Identified Strategies for a Mark II Containment (Continued)

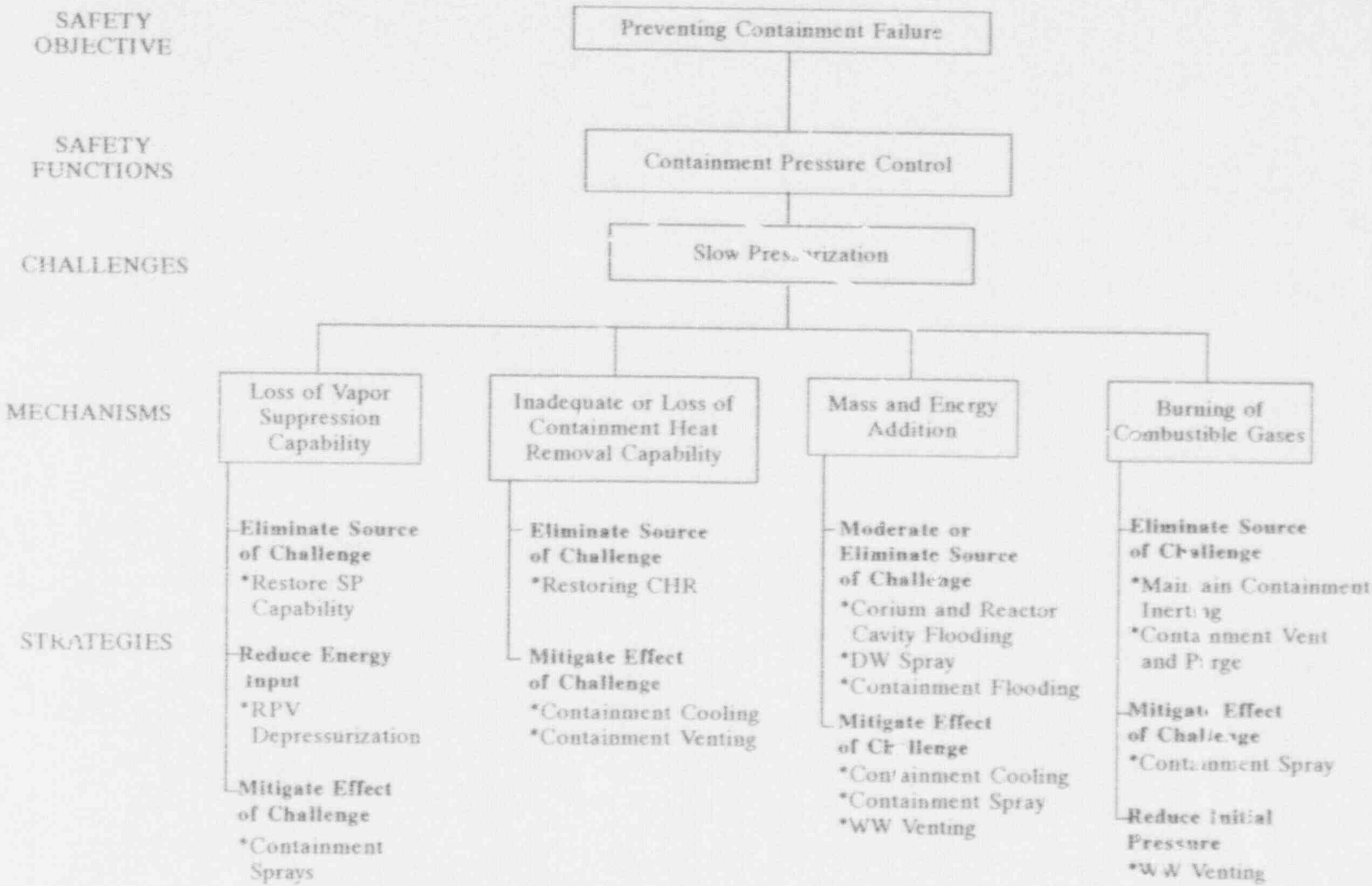

Figure 4.2 Safety/Objective Tree with Identified Strategies for a Mark II Containment (Continued)

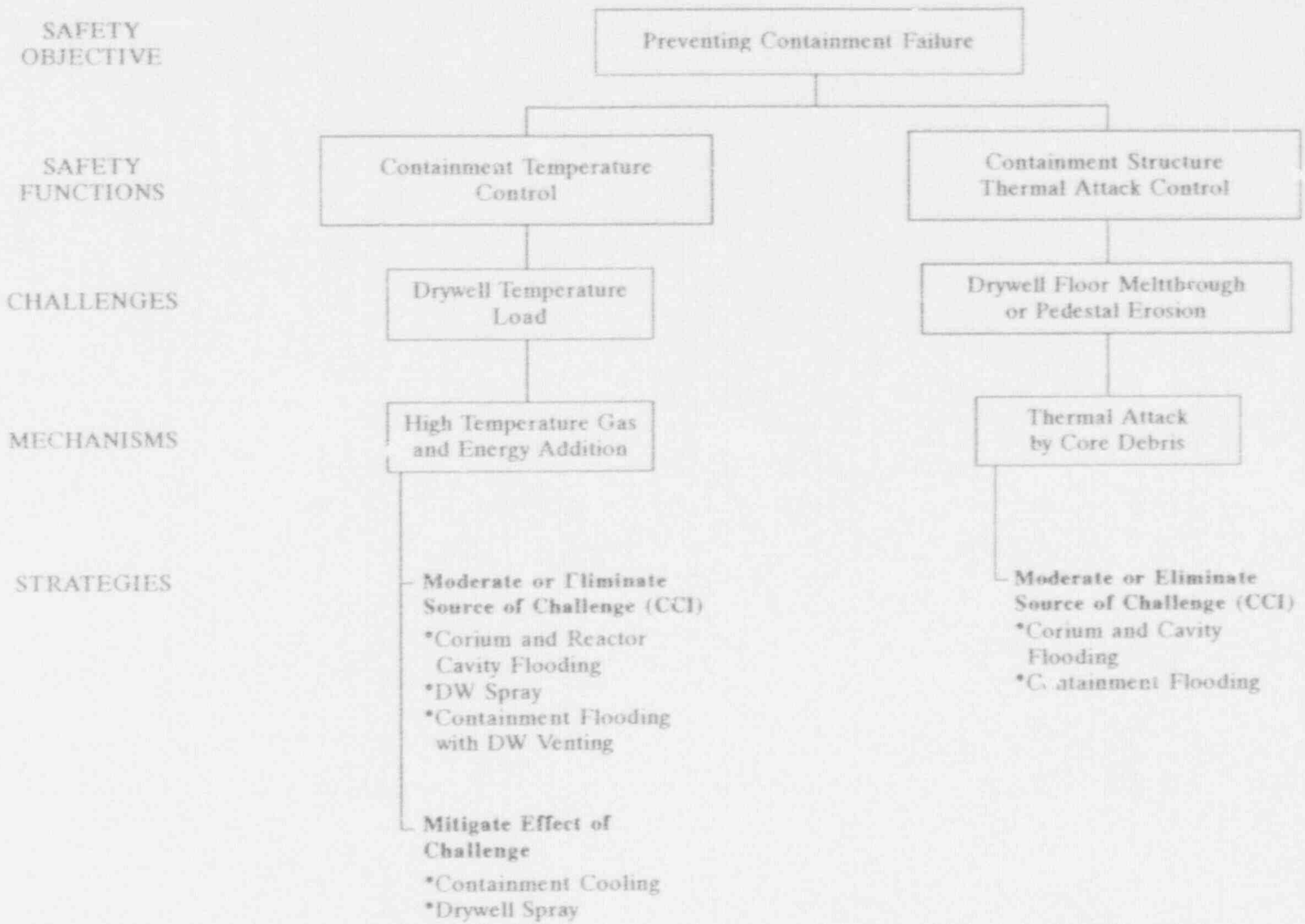

Figure 4.2 Safety/Objective Tree with Identified Strategies for a Mark II Containment (Continued)

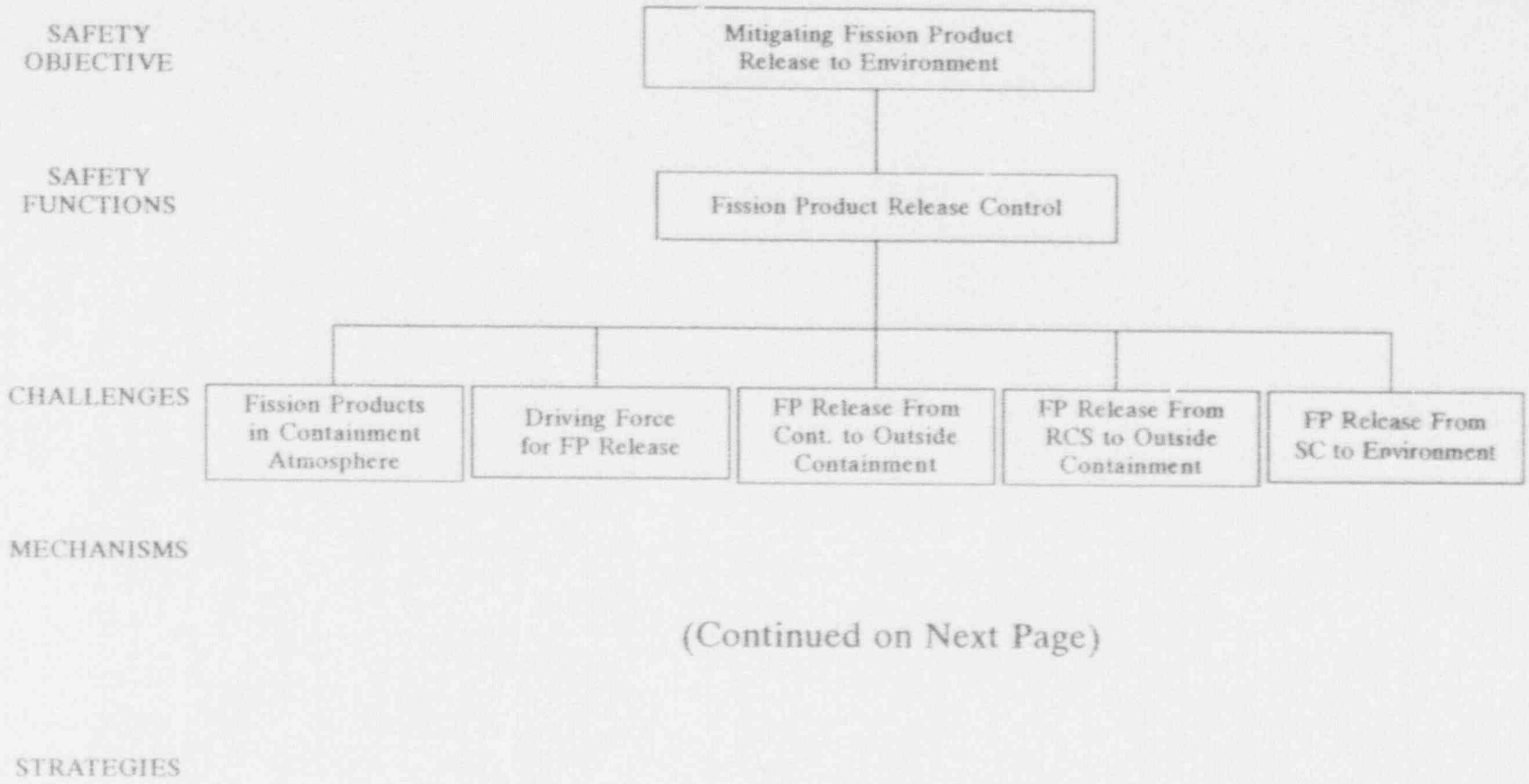

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Figure 4.2 Safety/Objective Tree with Identified Strategies for a Mark II Containment (Continued)

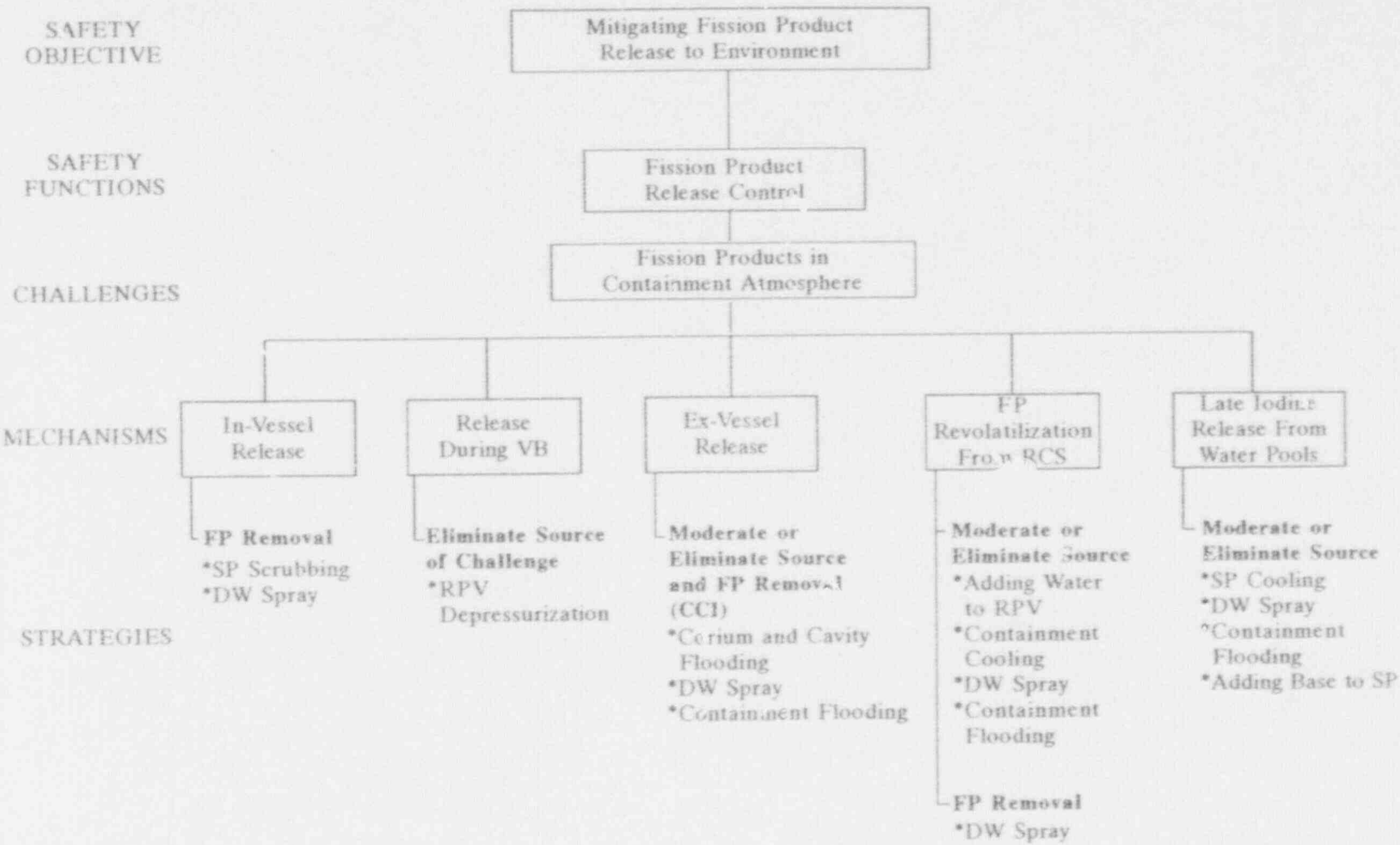

Figure 4.2 Safety/Objective Tree with Identified Strategies for a Mark II Containment (Continued)

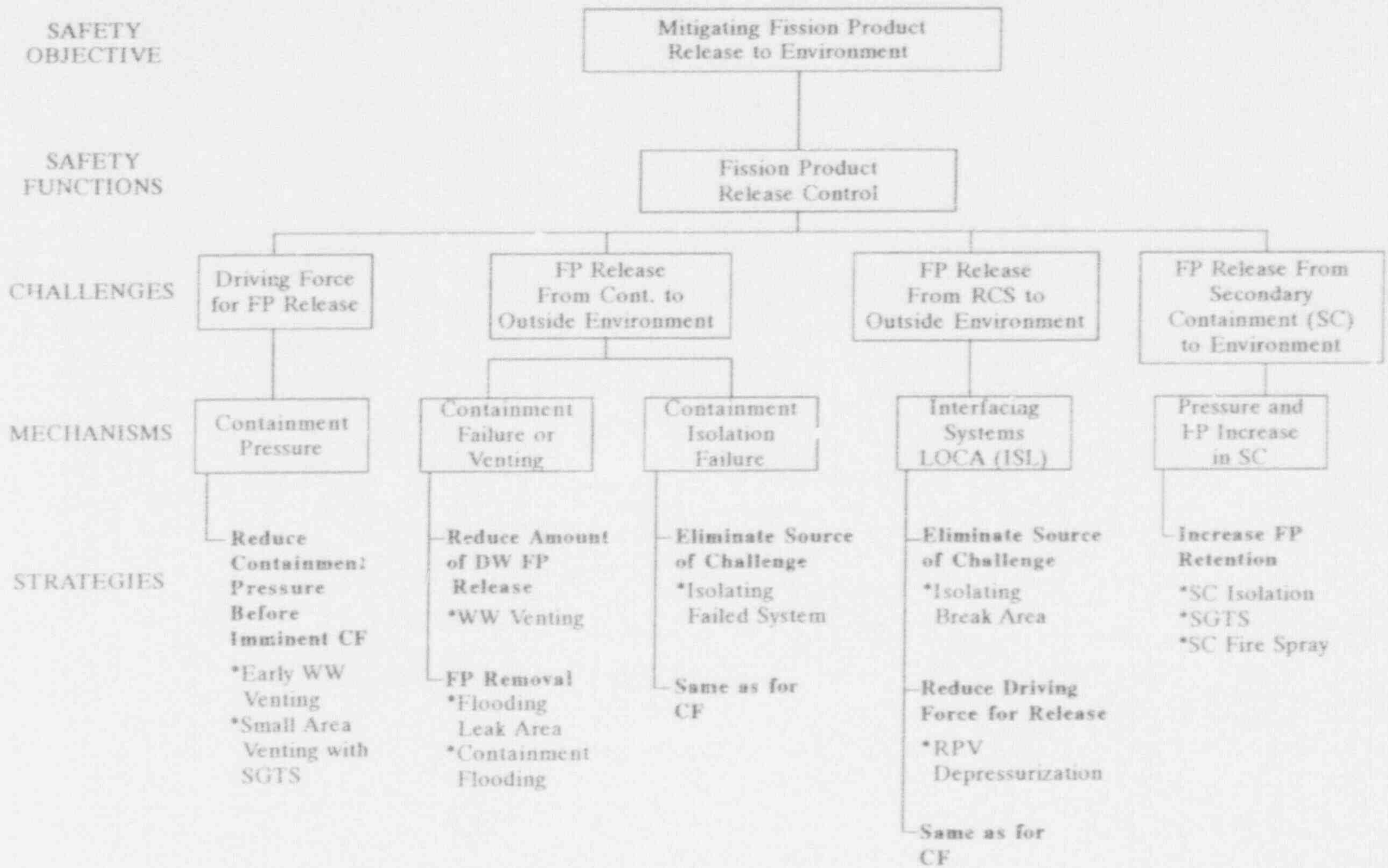

Figure 4.2 Safety/Objective Tree with Identified Strategies for a Mark II Containment (Continued)

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) [30]. 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 [27]. 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 [32]. 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 Limerick (and most likely for other Mark II plants also, see Table 4.2). 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 [7].

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 RHRSW, 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.

Strategy Discussion

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 inadequate NPSH, 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 Limerick 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 the Heat Capacity Temperature Limit (HCTL, about 150°F for RPV at system pressure), (2) the drywell and containment temperature cannot be maintained below their design temperature limits (340°F for the drywell and 220°F for the wetwell for Limerick), (3) the containment pressure cannot be maintained below the pressure suppression pressure (PSP), (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), or (5) the containment hydrogen concentration reaches 6% and the containment oxygen concentration is above 5%. 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. RPV pressurization may also recur when containment pressure is high enough so that the pneumatic supply pressure is insufficient to reopen the SRVs. The probability of maintaining the RPV depressurized can be improved by (1) extending the availability of dc power as discussed under resource management and (2) increasing the pressure of the pneumatic supply or maintaining a lower containment pressure.

As a result of the CPI program, the NRC staff has recommended to the commissioners an enhanced RPV depressurization system for Mark I containments [45]. This Mark I improvement is also recommended for consideration for the Mark II containments [46]. 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 fission products released to outside the containment during an ISL event. 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 in the secondary containment or the environment and a relatively low temperature and radiation level in the primary 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 compared to that of the downcomer vents, and the uncertainty of having intact SP vents at VB. On the downside, early RPV depressurization may accelerate the in-vessel core melt progression after the loss of core injection, could substantially increase clad oxidation, 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 melting 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)[1]. As indicated in Table 5.2, venting can also be useful for other reasons. These uses are: (1) to prevent containment pressure failure by reducing the base pressure before mechanisms that may cause rapid pressurization take effect and (2) to reduce the total amount of fission products released to the environment even after the loss of containment integrity. The adverse effects associated with venting, as shown in Table 5.2, are (1) the release of FP to the environment, (2) the drywell temperature load during CCI [7], and (3) the environmental loads on equipment in the SC. These issues of containment venting are discussed in the following.

Containment Venting to Prevent Containment Pressure Failure: 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 most effective 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 section 3.2.4.

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 Limerick is 70 psig, which is about 1.3 times the design pressure but far lower than the expected containment failure pressure of about 140 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 some BWR plant specific EOPs.

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 [30]. 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 Venting to Reduce Containment Pressure: The purpose of venting at a pressure lower than the PCPL is to reduce the initial containment pressure, in anticipation of a sudden, large pressure increase associated with a high pressure vessel breach, and thus prevent catastrophic early containment failure. Assuming containment failure is

[1]Wetwell venting is the preferred venting mode of containment venting and is the venting mode that must be used if the objective of venting is to reduce fission product release. Drywell venting is needed only in sequences where the required venting rate is high, e.g., ATWS sequences. The term venting is used in the following discussion to imply wetwell venting with the understanding that drywell venting will be included if wetwell venting itself is not sufficient to achieve the venting objective.

inevitable even with reduced initial containment pressure, such as might occur due to direct containment heating (DCH), this strategy may still have some benefit. It can reduce the total amount of fission products released to the environment because a part of the fission product inventory accumulated before vessel breach will be passed through the suppression pool and scrubbed.

This strategy requires knowing the current containment status as well as estimating the vessel breach time and the amount of the corresponding containment pressure increase. Therefore this strategy needs to rely heavily on previously established analytic models and predictions regarding the effect of vessel breach on containment loads under a variety of conditions. Because of the uncertainties in both predicting accident progression and knowing containment strength, the decision on early venting would most likely be based on a probabilistic approach. Section 4.2.5 has proposed a scheme to obtain data for making venting decisions.

The most important containment status variable for this strategy is the containment pressure. Indications from in-vessel instruments, such as the RPV pressure and core temperature, are needed to estimate the probability and timing of vessel breach and its impact on containment thermal and pressure loads. The offsite radioactivity release rate should be monitored during venting.

Early venting may also be needed for combustible gas control when the containment gas composition reaches the combustible limit. According to the BWR EPGs, venting and purging are to be used to control containment gas composition. Since the systems that can supply nitrogen to the containment have limited pressure capability, venting is required to reduce the containment pressure before these systems can be used (Section 3.2.1). Venting also reduces the initial containment pressure before combustion, and thus reduces the pressure load on the containment, should combustion occur. If combustion should occur during venting, releases from the containment would increase. However, the release would pass through the SP and thus benefit from pool scrubbing.

Wetwell Venting for Fission Product Scrubbing: A drywell leak is possible in some severe accidents. The driving force for fission product release is the pressure in the primary containment. Wetwell venting will reduce this driving force and provide fission product scrubbing through the suppression pool and thus reduce the total release of radionuclides to the environment. The need for this strategy can be inferred from the history of the accident progression (e.g., whether there is a core melt or a vessel breach) and the indications of the numerous radiation monitoring instruments in the secondary containment and offsite.

To avoid unnecessary venting there must be a clear indication that leakage from the drywell exists. There are normally a sufficient number of area radiation and temperature monitoring instruments in the secondary containment to determine the approximate location of the leak. However, the instruments in the secondary containment are generally qualified for environmental conditions much less severe than those in the containment. Therefore the conditions near the leak may be harsher than those for which the equipment is qualified.

The vent path should be closed after the leak area is isolated or reclosed. (The drywell leak area may be a function of containment pressure and temperature and may reclose on its own when containment conditions change.) The fission product release can be minimized by continuously monitoring the leak and making decisions accordingly. Previous analysis results regarding the most likely containment failure mode, such as those provided by the Containment Performance Report [17], can help in deciding on the initiation and termination of wetwell venting.

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) determining the vent paths to be opened (These depend on containment pressure rise rate. ATWS events require the opening of larger vent areas than other events.), (3) defeating the containment isolation valve interlocks (This step may need the assistance of an auxiliary operator to obtain the necessary equipment and make the needed temporary terminal connections.), and (4) opening the ac motorized valves from the control room. In the case of an SBO event ac power is not available and the auxiliary 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 wetwell venting for fission product scrubbing, as suggested here. If these two strategies are deemed to be important for a particular plant, they must be considered in choosing a disk rupture pressure, or some vent paths without rupture disks must be included.

In accordance with the requirements of NUREG-0696 [47], 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: Potential adverse effects that may result from venting have been discussed in Section 3.2.4. The important adverse effects include (1) loss of plant safety equipment due to containment depressurization and SP flashing, (2) SC contamination and resultant loss of function of safety related equipment or loss of accessibility to the secondary containment, (3) fission product release to the environment, and (4) hydrogen burning in an oxygen rich SC. Methods to avoid or mitigate these adverse effects are also discussed in Section 3.2.4. The use of a hardened vent path, as recommended by the CPI program, will further lessen the concern of SC contamination.

5.2.4 Strategies Related to Drywell Spray

As indicated in Table 5.2, drywell spray can be used to meet most of the challenges presented in Table 5.1. In addition to its designed function of containment pressure and temperature control, the drywell spray also can remove fission products from the containment atmosphere, provide water to the corium on the drywell floor, and reduce the probability of drywell floor melt-through, (Section 3.2.3). Furthermore, drywell spray also has a potential to mitigate the effects of hydrogen combustion (Section 4.2.2.1), to reduce the challenges associated with HPME (Section 4.2.2.2), and to prevent excessive SP boundary loads due to chugging (Section 4.2.1).

The use of drywell spray to control containment pressure and temperature under accident conditions is described in the BWR EPGs. Drywell spray is called for in the EPGs when the drywell temperature reaches the design temperature (or ADS qualification temperature) or when the containment pressure exceeds the suppression chamber spray initiation pressure (SCSIP). Drywell spray is also called for in the BWR EPGs when containment hydrogen and oxygen concentrations cannot be controlled to below predefined limits and containment pressure cannot be maintained below PCPL.

The use of the drywell spray as a water source to flood the reactor cavity and to add water to the corium on the floor will be discussed in Section 5.2.5

Drywell Spray for Fission Product Scrubbing: One of the most important functions of drywell spray in CRM is its ability to scrub fission products from the containment atmosphere (Section 3.2.3). This function is particularly vital after vessel breach when airborne fission product concentrations are high or when a containment leak or rupture exists and fission products are released without the benefit of SP scrubbing. The FP scrubbing capability of the drywell spray is more important for Mark II plants than for Mark I plants because of the higher SP bypass probability of the former (Section 4.2.3).

As a fission product scrubbing tool, drywell 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 reactor building or offsite. When operating the drywell spray, containment pressure and temperature should be constantly monitored to assure that the spray will not lead to a containment failure due to negative pressure (about -5 psig design). The possibility of deinerting the containment atmosphere by steam condensation should also be monitored when both H_2 and O_2 are present (Section 3.2.3). The current BWR EPGs provide guidelines for the derivation of a drywell spray initiation limit (DSIL), beyond which drywell spray should be prohibited to avoid the above potential adverse effects. Since the spray strategies discussed here can have different objectives and may be implemented under conditions significantly different from those when drywell spray is required by the EPGs, different drywell initiation limits may need to be established.

Drywell spray for fission product scrubbing is 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 drywell spray is uncertain. Alternate means of obtaining necessary indications may have to be planned in advance.

Drywell Spray as a Heat Sink During RPV Blowdown: Drywell spray, activated before vessel breach and operating during the course of a RPV blowdown, has the potential to mitigate the effect of the challenges associated with HPME by providing an additional heat sink for the blowdown gases and the core debris dispersing into the containment atmosphere (Section 4.2.2.2). The effectiveness of the spray as a heat sink under these conditions has not been analyzed, but would depend on the droplet size and spacial distribution of the spray and its flow rate as well as on the debris size and dispersion rate. Further research is required in this area. Prediction of droplet size may be difficult because of equipment congestion in the drywell. Equipment in the spray path may result in smaller droplets due to breakup, or in larger droplets which have coalesced and drip from equipment surfaces.

Operator Actions and Equipment Requirements: Once the decision to use the drywell 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 drywell spray limits during spray operation to assure that spraying will not cause unacceptable adverse effects.

In using drywell spray to meet a variety of challenges occurring at different phases of an accident with very different containment conditions there are also limiting conditions that restrict the use of drywell spray. The operating staff also needs to have guidance available to determine the appropriate initiation time and adjust the flow rate of the spray. In addition, they need to decide whether to add water to the vessel or to use the containment spray, if both systems are competing for the same water supply.[2] Clearly defined procedures or guidelines are needed to avoid confusion in the management of drywell spray. Such procedures currently exist in the BWR EPGs. However, modification to existing guidelines or additional guidelines may be required because of the extended applications of drywell spray in CRM.[3]

Potential Adverse Effects: Potential adverse effects of drywell spray have been discussed in Section 3.2.3. The primary concerns are the negative pressure load, containment deinerting, and a possible containment pressure surge due to steam generation from corium-water interaction. If the containment integrity has been lost, a pressure surge in the containment would cause additional discharge of the containment atmosphere and FP release. The timing of the spray needs to be coordinated with other severe accident management activities, such as offsite evacuation, to minimize the offsite risk. The rate of the spray should also be carefully assessed: Too high a

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[2]The competition of spray water with vessel injection is a valid question even after vessel breach because some of the reactor core materials will remain in the vessel. Choosing the best water application method will be a problem if only a single limited water supply system is available. One solution to this problem is an arrangement that can supply water to both core injection and containment spray simultaneously.

[3]In some cases, the existing spray restriction conditions, which apply during the very early phase of an accident, may need to be removed to achieve better CRM results. For example, existing EPGs prohibit the use of the drywell spray when the vacuum breakers between the drywell and the wetwell are flooded in order to prevent an unacceptable pressure differential between these two compartments. This may restrict the use of the drywell spray for certain severe accident scenarios which include the continuous production of large amount of noncondensible gases in the drywell.

rate may generate an unacceptable pressure differential between the drywell and the wetwell. A less than sufficient spray flow rate may result in an additional pressure increase from steam generation (Section 4.2.2.2).

5.2.5 Reactor Cavity Flooding

This strategy involves two 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. (However, current studies indicate that small amounts of water in the cavity may enhance DCH pressurization [48].) The second is to continuously add water to the core debris after it falls into the reactor pedestal region and interacts with concrete, to: (a) moderate or terminate the progress of CCI, (b) reduce the probability of a suppression pool bypass due to the breach of downcomer vents or drywell floor drains, and (c) provide an overlying water pool for fission product scrubbing. The effects of water on the phenomena associated with HPME and CCI still involve some uncertainty, but are expected to be beneficial and desirable. This strategy is not applicable to the Mark II plants that have downcomers inside the reactor pedestal region. Unless these downcomers become blocked during the course of the accident, most of the core debris discharged from the vessel in these plants will be directed to the suppression pool and CCI in the reactor cavity will be limited (Section 4.2.3).

Most of the time, only indirect inferences are available to deduce the existence of the challenges that are addressed by this strategy (Table 5.1). For example, to assure sufficient water in the reactor cavity before vessel breach, the drywell spray needs 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 the time and flow rate of spray operation.

Operator Actions and Equipment Requirements: Drywell spray is the only means to add water to the reactor cavity before vessel breach, other than flooding up the containment. The operator actions and equipment required are the same as those discussed in Section 5.2.4 for drywell spray. However, after reactor breach, if both systems are operational water can be added to the corium either through the vessel by the use of core injection or by drywell spray. While adding water through the vessel can keep the core materials in the vessel cooled (Section 4.2.3 on revolatilization from RCS), adding water via the drywell spray can provide an additional fission product scrubbing capability and also cool the containment atmosphere. For the Mark II containments that have a shallow reactor cavity, adding water via the drywell spray may also reduce the probability of downcomer melt-through and thus reduce the probability of SP bypass. On the other hand, for the Mark II containments that have a deep reactor cavity, adding water via the drywell spray may be impossible after VB. Water can be added only through the RPV break area in these plants (Section 4.2.3).

Since this strategy is likely to be implemented when significant uncertainties in plant conditions exist, it will be more difficult to provide specific procedures and predefined parameter values for strategy implementation. It may be more desirable to provide flexible guidelines as well as relevant data and calculational tools, as discussed in Section 3.3.2.4. As discussed above, successful implementation of this strategy will require coordination with other accident management activities.

Potential Adverse Effects: The same adverse effects associated with the drywell spray strategy discussed previously are of concern here. In addition, the existence of water in the reactor cavity before vessel breach may result in FCI at vessel breach (Section 4.2.2.2). This pressure increase due to FCI, when combined with the increase from mass and energy addition at vessel breach, may threaten containment integrity. However, this adverse effect has to be weighted against the possible beneficial effects for the mitigation of HPME and CCI. Current knowledge indicates that flooding the reactor pedestal region and drywell floor is more desirable than maintaining these regions dry, although phenomenological uncertainties are high.

Adding water could also result in a puff release of fission products to the environment if it occurs during containment venting or after containment failure. The decision to flood the corium may also need to be coordinated with other accident management activities, e.g., the time of venting and offsite evacuation, if these are needed.

5.2.6 Combustible Gas Control

The amount of combustible gases that can be generated from both in-vessel core degradation and CCI is significant and may create an atmosphere that exceeds combustible limits (Sections 4.2.2.1 and 4.2.3). (Note that there may also be enhanced radiolysis due to the corium's location and its exposure to water.) Combustible gas control is primarily achieved by maintaining an inerted containment, i.e. by avoiding the introduction of oxygen. If the containment gas composition reaches unacceptable limits, inerting can be regained by a containment vent and purge operation. Containment spray can also be used to mitigate the effect of combustion if the above efforts fail (Section 4.2.2.1). The hydrogen recombiners if still functional, can also be used to achieve some limited control.

Instrument indications are available for hydrogen and oxygen concentrations in both the wetwell and the drywell. Samples of containment atmosphere can also be obtained and analyzed during an accident to provide data on containment gas concentration. Detailed knowledge of hydrogen stratification is important. However, existing hydrogen monitoring system in operating plants may not be capable of providing such information.

If wetwell venting or drywell spray is needed for combustible gas control, the requirements for the operation of those systems need to be followed. The pressure and flow limitations of the systems that can supply nitrogen to the containment, i.e., CAC system, need to be observed to avoid damage to these systems (Section 3.2.1). If combustion occurs during wetwell venting the discharge of the containment atmosphere to the outside may increase significantly.

5.2.7 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 (Section 4.2.3).

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.

One of the important leak areas is the drywell head area. Flooding this area will have a significant effect on FP release and offsite risk (Section 4.2.4). Flooding this area will also provide an external cooling to the drywell head seal and thus may prevent its failure due to thermal load, and may reduce or even prevent the leak [12].

As discussed in the Mark I report [7], even a moderate leak area would result in a significant volumetric flow rate, 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.

5.2.8 Primary Containment Flooding

Primary containment flooding, once achieved, can provide cooling to the corium and FP scrubbing with minimal use of active systems. 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. This strategy could also be of benefit for containment and release control.

Even with the RPV failed, flooding the containment to the TAF level will (1) provide water to the core material remaining in the vessel and thus reduce fission product release from revolatization and (2) provide water to the corium on the drywell floor and thus terminate, or slow down the rate of, CCI. The second objective is more important for the Mark II containments that have a deep reactor cavity, because, for these containments, it is more difficult to add water to the reactor cavity, and CCI, once started, is likely to continue until the drywell floor is burned through. In addition to the above objectives, 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.

The amount of water required, the systems that can be used, the time needed,and other important considerations for containment flooding have been discussed in Section 4.2.3.

Once this strategy is implemented, the ability to perform other CRM strategies becomes very limited. However, this strategy may be very desirable after the vessel is breached because (1) it involves a minimal use of active equipment and (2) its beneficial effects are equivalent to those of many other strategies discussed in this section. 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. It also reduces the rate of CCI and improves the probability of terminating CCI.

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 (i.e., pressure rise per unit energy input to the containment atmosphere), and increase the hydrostatic load on the containment. Even if the mass and energy generated from CCI is terminated, the energy from the decay heat will raise containment pressure steadily and drywell venting may be required to remove the added energy. Therefore, it is important that all drywell vent paths are not flooded and remain operational for the duration of the accident.

As discussed in Section 4.2.3, there is a time period, after the wetwell vent paths are flooded but before the corium on the drywell floor is completely covered with water, during which releases from the containment may not be scrubbed. Preplanning is needed to ensure that containment venting is not required or that drywell sprays are available during this time. 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.

5.2.9 Strategies Related to SC Fission Product Retention

In the event of primary containment failure, fission products are usually discharged to the secondary containment before they pass into the environment. Therefore the secondary containment provides the last opportunity to mitigate the release of radioactive materials to the environment. Retention of fission products in the secondary containment can be achieved by (1) natural deposition on secondary containment structures and cooling coils within heat exchangers, (2) the operation of the Standby Gas Treatment System (SGTS), and (3) the operation of the fire spray. These systems and their ability to retain fission products were described in Section 3.1.3 and their roles in CRM were discussed in Section 4.2.4. Additional discussion is provided below.

<u>Using the SGTS to Reduce Fission Product Release</u>: The BWR EPGs call for the secondary containment 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, i.e., the off-gas stack.

The designed discharge capacity of the SGTS is small when compared with the expected containment leak rates (Section 4.2.3). 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 SC 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. The flow through the SGTS, and thus the benefit of filtering, could be enhanced if the SGTS can be operated in a recirculation mode. 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 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 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 SC pressure is low (i.e., a low containment leak rate).

Additional issues regarding the SGTS were discussed in Sections 3.1.3.2 and 4.2.3

<u>Secondary Containment Fire Spray for Fission Product Retention</u>: 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 Limerick has a diesel-driven pump as a backup to an electric motor-driven pump. Each pump has a capacity of 2,500 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 near the containment break may be harsher than that for which the equipment is qualified.

On the negative side, 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 use 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

Challenge	Parameters: Direct	Parameters: 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 Levels	
10. FP From Containment to Outside	Secondary Containment (SC) or Environment Radiation Levels	Containment Radiation Levels
11. FP From RCS to Outside	SC and Environment Radiation Levels	Containment Radiation Levels
12. FP From SC to Environment	SC and Environment Radiation Levels	

Table 5.2 Strategies and the Challenges Addressed by the Strategies

		Challenge (See Table 5.1 for Description)											
	Strategy	Loss of Resources	SP Dynamic Loads	Containment Pressure	Drywell Temperature	Ex-vessel Steam Explosion	Rapid Containment Pressurization Related to HPME	Burn of Combustible Gases	Core-Concrete Interaction (CCI)	FP in Containment Atmosphere	FP From Containment to SC	FP From RCS to SC	FP From SC to Environment
1	Resource Management	+(1)											
2	RPV Depressurization		+				+			+		+	
3	Wetwell Venting			+	-(2)	+	+	+			-		
4	Drywell Spray		+	+	+	±(3)	+	±	+	+	+		
5	Reactor Cavity Flooding					-	+		+	+			
6	Combustible Gas Control							+					
7	Leak Area Flooding										+		
8	Primary Containment Flooding								+	+	+		
9	SC FP Retention												+
10	Other Strategies												
	Containment Cooling			+	+					+			
	SP Cooling	+	+							+			
	SP Water Level Control	+	+										
	Isolate Leak Area										+	+	

Note: (1) + Mitigating Effect
(2) - Potential Adverse Effect
(3) ± Uncertainty or Secondary Effect

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 Limerick Generating Station was used as the surrogate plant for this assessment. 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 on-site diesel generators. For the Mark II plants that utilize a BWR/5 reactor, this includes the loss of the dedicated diesel generator for the High Pressure Core Spray (HPCS) system. An SBO leads in a Mark II BWR to the loss of all active engineered safety features except the steam turbine powered Reactor Core Isolation Cooling (RCIC) system, and, for the plants that utilizes a BWR/4 reactor the High Pressure Core Injection (HPCI) system (Section 3.1). Since both the RCIC and the HPCI system require dc power for control, they fail after the depletion of station batteries. The RCIC and HPCI turbines may also trip because of high turbine exhaust pressure (i.e., containment pressure) or high SP temperature, both of which occur due to the loss of containment heat removal in SBO. 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 to two hours 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 and the HPCI system, in addition to the loss of all ac power. 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 (Table 4.3).

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

Figures 6.1 and 6.2 present containment pressure and temperature histories of a fast SBO sequence calculated by the BWR-LTAS, BWRSAR, and MELCOR codes for a synthetic BWR/4 Mark II plant [13]. The synthetic plant uses reactor and containment parameter values of Susquehanna (which are similar to Limerick) and has a deep in-pedestal reactor cavity, typical for BWR/5 plants (Section 3.1.1.3). The reactor was not depressurized in the calculation. This could be due to the loss of dc power, which is required for ADS actuation.

In the fast SBO case discussed above, the core starts to melt at about two hours, and the RPV is breached at about four hours after accident initiation. The mass and energy addition to the containment from the high pressure RPV blowdown causes a sharp pressure rise in the containment immediately after VB. Containment pressure continues to rise after VB, primarily due to the energy and noncondensible gases released from CCI, until the containment failure pressure of 150 psia is reached at about 10 hours after accident initiation. Since the calculation did not incorporate a containment failure model, Figure 6.1 shows a continuous containment pressure rise after the containment failure pressure is reached. Containment pressure reaches approximately 175 psia at the end of the calculation, when the drywell floor (4.7 ft thick) is penetrated by CCI.

As shown in Figure 6.2, there is a significant temperature stratification in the containment airspace. The containment temperature basically follows the same trend as the containment pressure, except for a temperature spike occurring immediately after VB. The containment temperature reaches about 800°F when containment failure pressure is reached, and continues to rise afterward. Unlike the containment pressure, which would decrease after the containment fails, containment temperature would continue to rise after containment failure.

In a slow SBO sequence, the containment pressure before core degradation depends on the duration of core injection. Figures 6.3 to 6.5 present the containment pressure, containment temperature, and suppression pool water temperature time histories, respectively, of a slow SBO sequence for the synthetic Mark II plant. Core injection was assumed to be available until battery depletion, six hours after accident initiation. According to the calculation, the suppression pool is not saturated, and the containment pressure is low (about 10 psig) before core degradation (about 12 hours after accident initiation). However, the suppression pool can be saturated and the containment pressure may be significant if core injection is maintained for a longer time duration, either by an extended battery life or by the use of an alternate water supply. In the extreme case, when core injection is maintained indefinitely, the SBO sequence would behave like a TW (loss of CHR) sequence, with the removal of containment heat a major concern.

RPV depressurization was assumed to be successful in the slow SBO sequence. The RPV pressure was controlled at about 200 psia in the calculation by the use of the HPCI system in the test mode. (RPV depressurization could also be controlled by the SRVs, and it would be the method used in BWR/5 plants which do not have the HPCI system.) However, control of the RPV pressure is lost after battery depletion, and RPV pressure increases until the vessel is breached at high pressure. The containment pressure and temperature responses after battery depletion in the slow SBO case are in general similar to those in the fast SBO case.

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 above 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 presented in Figures 6.1 and 6.2. Corresponding information for the slow SBO sequence, based on results presented in Figures 6.3 and 6.4, is shown in Table 2. Except for the difference in the timing of major events the two sequences present similar challenges and thus require similar strategies. 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 would be declared at the beginning of a slow SBO sequence, and with the additional loss of dc power, 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 (Section 3.3.1) and entry into the emergency guidelines. 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 the accident. Steam generated in the RPV from decay heat is discharged through the SRVs, or the RCIC or HPCI turbine exhaust, to the suppression pool 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.

As shown in Tables 6.1 and 6.2, core melt starts at about 2 hours after accident initiation for the fast SBO case and about 6 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 pressure before VB is about 30 psig for the fast SBO case and 50 psig for the slow SBO case. The discharge of the hot hydrogen gas to the containment also causes a temperature increase in the containment. The containment temperature can be high in some locations but is in general less than 400°F. The challenges associated with these conditions are not significant and actions are unlikely because of the lack of electric power.

The amount of hydrogen released to the containment atmosphere is in general sufficient to cause the containment hydrogen concentration to reach combustible, or even detonatable, limits before vessel breach (Section 4.2.2.1). Since a Mark II containment is inerted, a hydrogen combustion is not expected to occur. The most important challenges to containment integrity during this time are those associated with HPME, occurring at the end of this phase. Since MELCOR does not have a DCH model, the pressure rise due to HPME shown in Figures 6.1 and 6.3, although already significant, does not include the effect of DCH, and a greater pressure rise would result should DCH occur. Maintaining the RPV depressurized is therefore, very important. A depressurized RPV not only mitigates the challenges associated with HPME, but also eliminates the suppression pool boundary loads (Sections 4.2.1 and 4.2.2), and changes the core melt process and the mode of debris ejected from the vessel [13]. BWRSAR/MELCOR analyses of fast SBO sequences showed that there would be no significant containment pressure rise at VB if RPV pressure was low before VB, and the containment pressure after VB would be much lower in the low pressure case than in the high pressure case (40 psig versus 80 psig). It is noted that the timing (or reactor water level) for RPV depressurization will affect the metal-steam reaction (an exothermic reaction) in the vessel and the timing of vessel failure.

Strategy Application

In some cases early wetwell venting can be used to reduce the base pressure and thus the impact of HPME (Section 5.2.3). Early venting is advisable only if the vent path can be reclosed before significant fission products are released. Venting the containment atmosphere before core degradation begins is therefore more desirable because the radioactivity in the containment atmosphere is low. However, according to BWRSAR/MELCOR calculations, the containment pressure before core melt (CM) is too low (about 10 psig for the slow SBO case and less for the fast SBO case) for this strategy to be practical. Venting after CM but before VB, while it will result in some FP release, will have SP scrubbing, and result in less FP release than an early HPME induced containment failure. The containment pressure before vessel breach, as shown in Figures 6.1 and 6.3, reaches its highest value after the second major SRV discharge, about one hour before VB, and venting during this time period will give the greatest base pressure reduction. Early venting therefore has the potential to reduce the containment base pressure by 30 psi for the fast SBO case and 50 psi for the slow SBO case. Pressure reductions of these magnitudes may be helpful to prevent an early containment failure immediately after VB. The containment pressure reduction by early venting will be more significant if the pressure before CM or VB is greater, as would happen in the cases where high pressure injection can be maintained for a longer time. However, the benefit is uncertain because of the uncertainties in containment pressure and containment pressure capability predictions. The ability to vent the containment during an SBO sequence is another uncertain issue and has been discussed in Section 3.2.4.2.

It should be noted that early venting may not be beneficial for risk reduction. In fact, it may cause unnecessary FP release if the containment integrity can be maintained without containment venting, or change a late release case (due to late CF) to an early release case. A simplified probabilistic risk analysis showed that preemptive venting, i.e., venting before a significant rise in containment pressure in a SBO sequence to prevent containment failure, would result in significant increases in the important risk measures considered in the analysis (i.e., the population dose, the mean number of latent fatalities, and the offsite cost) [12]. However, in the analysis, it was assumed that the vent path, once opened, could not be reclosed. The risk will certainly be reduced if the vent path can be reclosed before significant fission products are released to the containment atmosphere. Additional considerations using actual plant data during an accident, such as those discussed in Section 4.2.5, will also help to make an optimum venting decision.

There are other strategies that may be useful during this phase of a SBO sequence. As discussed in Section 6, flooding of the reactor cavity before VB may have a beneficial effect on the load associated with HPME. The operation of drywell spray, in addition to providing water to the reactor cavity, may also mitigate the effects of HPME. However, it is unlikely that water would be available for the drywell spray at this time. Available water would be used for core cooling before vessel breach. However, in sequences where the RPV is repressurized after the loss of dc power, an alternate water supply system, e.g. the fire water system, may not have sufficient head to deliver water to the RPV, consequently, it could be used for the drywell spray.

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., the pressure suppression pressure, PSP, or the primary containment pressure limit, PCPL), but some of the actions required are not relevant any more (e.g., RPV depressurization). Containment cooling and drywell 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 Figures 6.1 and 6.3, the containment venting pressure (PCPL, 70 psig for Limerick) is reached 4 hours after accident initiation for the fast SBO sequence and about 16 hours after accident initiation for the slow SBO sequence; and containment failure pressure (135 psig for Limerick) is reached about 10 and 21 hours, and drywell floor penetration (by CCI) occurs about 12 and 25 hours, after accident initiation, for the fast and slow SBO sequences respectively.

CCI is the most important mechanism for containment loading in this phase of the accident. 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 an erosion of the reactor pedestal and the drywell floor. The hot core debris may also attack the downcomers or the drywell drains and thus cause a SP bypass shortly after VB (about 20 minutes, Section 4.2.3). The strategies to cool the core debris and control the progression of CCI are therefore important (section 4.2.3). Flooding the reactor cavity before vessel breach and continuously adding water to the corium increases the probability of slowing down the progress of, or even terminating, CCI. Water can be added to the reactor cavity by drywell spray before VB, or to the hot corium, either through the RPV break area by core injection, or through the drywell by drywell spray, after VB. Drywell spray, in addition to providing water to the corium, can also be used to control containment pressure and temperature loads.

The effect of drywell spray on drywell temperature is particularly important. Even a moderate spray flow rate can prevent a significant drywell temperature load [7]. Conversely, drywell spray does not have a significant effect on containment pressure response if the flow rate is low (e.g., 500 gpm), but could delay the time to containment pressure failure significantly (by a few hours) if the flow rate is high (e.g.,6,000 gpm) [7]. However, a high spray flow rate is not likely to be available during a SBO sequence. The most likely water supply system available at this time is the fire water system (Section 3.2.1), which can supply a flow rate of about 570 gpm at zero psid and has zero flow at 82 psid [13]. A MELCOR analysis using such a drywell spray was performed for a fast SBO sequence with RPV depressurization to evaluate the effect of drywell spray [13]. The drywell spray was initiated at the time of VB in the analysis, although, according to the BWR EPGs, if available, it could have been activated earlier. The water that can be accumulated on the drywell floor depends on the height of the downcomer lip above the drywell floor, which is about 1.5 ft for the Mark II plants that do not have downcomers inside the pedestal region. In the analysis, the spray was terminated 7.5 hours after its initiation due to high containment pressure (exceeding the pump cutoff pressure). After spray termination, it took another four hours to boil off the water pool on the drywell floor.

The effect of drywell spray on the attack of the drywell floor is shown in Figure 6.6 (from the calculation of Reference 13). The availability of water to the corium significantly delays drywell floor burn-through even for the deep cavity model used in the analysis. Debris spreading for some Mark II plants would result in greater surface area for heat transfer between water and corium (Section 3.1.1.3) and probably less severe concrete attack. The drywell spray also reduces containment temperature. The containment temperature is limited by the drywell spray to below about 600°F for all the containment regions used in the calculation. The containment pressure is also slightly lower for the spray case than the no-spray case at the time of spray termination. However, the containment pressure at drywell burn-through is greater for the spray case (135 psig in the drywell, 16.5 hours after accident initiation) than the no-spray case (105 psig, 13.5 hours). This is due to the decrease of containment free volume by containment water addition from drywell spray, and the continuous boiling of water overlying the corium [13].

PCPL is reached during this time phase for all of the SBO cases discussed above. However, wetwell venting may not be carried out because of lack of electric power. Even if containment venting with SP scrubbing is available, it may not be advisable to vent the containment at this time because of the high radioactivity in the containment atmosphere. A simplified probabilistic risk analysis indicates that venting at PCPL via a hardened wetwell vent does not reduce SBO risk because the containment may not fail in some of the venting cases [12]. In fact, risk may actually be increased because there is a high probability of SP bypass in Mark II containments [12]. On the other hand, if containment failure is certain, wetwell venting will in general cause a risk reduction [14]. The decision to vent the containment should therefore depend on projections of the risks associated with (and without) containment venting, based on existing on-site and off-site conditions during the accident (Section 4.2.5). There is another potential adverse effect that needs to be considered when making the containment venting decision. The containment temperature load may be more severe if the containment is vented during CCI (see Section 4.2.3 for more detailed discussion). This is because the gases removed from the containment atmosphere via venting have a lower temperature than the gases released from CCI. This indicates that venting, without accompanying containment cooling, e.g. by drywell spray, would result in a more severe temperature load and potentially an earlier containment failure due to the combined temperature and pressure load [7].

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.

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 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 containment integrity.

If containment failure is inevitable, actions, if feasible, should be taken before containment failure to reduce the potential for FP release after containment failure. Actions that reduce the amount of fission products in the containment atmosphere and the driving force of FP release (i.e., containment pressure) will be more effective if they are carried out when the containment is still intact (than if they are carried out after the containment has already failed). Strategies that can be useful for FP release control have been discussed in detail in Section 4.2.2, and individual strategies have been discussed in Section 5. A general 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.

As shown in Tables 6.1 and 6.2, without operator actions, the primary containment would fail in about four to five hours after vessel breach. If the containment fails in the drywell, the release of the fission products would be directed to the outside of the primary containment, bypassing the suppression pool. This scenario has been used in the Mark I report as the base case to discuss the effects of a few strategies on FP release. As discussed in the Mark I report, the objective of such strategies is to change the release condition from one represented by an accident progression bin (APB) with a greater FP release potential to one with less. Using a partial release CRET, the Mark I report has addressed the effects of a few strategies where FP release data are available from NUREG-1150 APET analysis. Although the Mark I results may not be completely applicable to the Mark II plants, they do provide adequate estimates of the effects of the various strategies for a Mark II plant. The Mark I results indicate that, when compared with the base containment failure case, both drywell spray and wetwell venting are effective in reducing the release of fission products. On the other hand, an early drywell leak directly into the reactor building, such as occurs in a drywell head leak case, would result in a FP release comparable to that of the base case and is significant. This shows the importance of flooding the leak area if the drywell spray is not available.

The strategy of flooding the drywell head has been discussed in Sections 4.2.4 and 5.2.7. More detailed discussion can also be found in Reference 12. Simplified probabilistic risk analyses have been performed in Reference 12 to evaluate the effect of deliberate flooding the drywell head for SBO sequences. Results showed that drywell head flooding would slightly reduce the offsite consequences when compared with a case without drywell head flooding. It results in a 3.9% reduction in mean latent fatalities, a 3.7% reduction in mean population dose, and a 7.3% in mean offsite cost. It should be noted that these reductions are from a sensitivity analysis based on probabilistic risk analyses, and do not represent the effect of flooding the drywell head (versus not flooding drywell head) in a drywell head failure case, because included in the comparison are many other containment failure modes that do not involve the failure of the drywell head. A direct comparison of the consequences from the releases of a drywell head failure, both with and without drywell head flooding, will certainly show a more significant effect of drywell head flooding on FP release and offsite consequence.

As discussed above, wetwell venting and drywell spray are important strategies to mitigate the consequence of a severe accident. Both of them have multiple beneficial effects on accident mitigation. The effects of these strategies have been investigated in Reference 14. The containment model used in Reference 14 was based on the Limerick plant. Because of the shallow reactor cavity of the Limerick plant, the corium was assumed to spread to an area of 5 meters (the radius of the inside of the reactor pedestal region is 3 meters). The sequence analyzed was a TQUV sequence which assumed the loss of all core injection at the beginning of the accident and with a successful RPV depressurization. (This TQUV sequence is similar to the fast SBO sequence discussed above in Section 6.2.3.2.) The analysis was performed using the Source Term Code Package (STCP). When compared with the results calculated by BWRSAR/MELCOR (Section 6.2.3.2), the STCP predicted a greater containment pressure rise after VB and a containment pressure failure at an earlier time (145 psia failure pressure at 8.5 hours).

The cases analyzed in Reference 14 are shown in Figure 6.7. In addition to the base case, where the containment was assumed to fail in the drywell by overpressure, two venting cases and two spray (with venting) cases were also investigated. In the first venting case (VT1), the wetwell vent path (the 6-inch ILRT line, Section 3.2.4) was opened w[illegible] the containment pressure reached 70 psig (PCPL for Limerick) and remained open for the duration of the acc[illegible]. In the second venting case, the vent path was reclosed when the containment pressure dropped to 55 psig, and reopened when the pressure reached PCPL again. This resulted in an intermittent valve opening and closing operation. This intermittent wetwell venting operation was retained in both spray cases, with spray flow rate[illegible] of 250 gpm and 500 gpm respectively.

The consequences of the above five cases, obtained by the MELCOR Accident Consequence Code System (MACCS) and using the Peach Bottom meteorological data, are presented in Table 6.3. The benefit of wetwell venting and drywell spray are clearly demonstrated in Table 6.3. Significant benefits are also realized in the reduction of the decontaminated and interdicted farm areas. The interdicted farm area is reduced from 272 square miles for the nonmitigated base case to 1 square mile for case SV1. Not reflected in the consequence comparison is the effect of drywell spray on drywell temperature. While the average drywell temperature after the initiation of wetwell venting for the venting case was about 1,000°F, the drywell temperature for the spray case was maintained at below 300°F through almost the entire transient (except for a short period when the drywell temperature rose to about 400°F). The drywell could therefore fail in the venting case from the temperature load, or a combination of temperature and pressure loads, and as a consequence, the release in the venting case would be without the benefit of SP scrubbing, and more severe consequence would result. The drywell spray, in addition to its ability to maintain a lower drywell temperature, is also an important tool to remove fission products from the containment atmosphere. This FP scrubbing capability is particularly important if there is a suppression pool bypass, which has a high probability for Mark II containments (section 3.1.1.3). Furthermore, If there is a SP bypass, the mitigating effect of wetwell venting on FP release is lost because the release will not have the benefit of SP 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 or the containment, 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 while the reactor failed to scram. 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 SP would lead to a rapid heat up of the pool and containment pressure rise. The containment would fail if recovery or mitigative actions are not successful. The flashing of the SP water would cause the ECCS pumps to fail by cavitation in some Mark II plants, and core melt, then vessel breach, would follow in these plant. In the Mark II plants where the ECCS pumps have been designed to pump saturated water, the pumps may not be lost at containment failure. Nevertheless, core injection may still be lost because of random hardware faults or operator errors (e.g., failure to depressurize the RPV).

In a slow ATWS sequence, core injection is lost late so that core damage, core melt, and vessel breach occur after containment failure. In a fast ATWS sequence, core injection is lost early such that core damage occurs in the short term, and core melt and vessel breach occur prior to containment failure. This may occur if the high pressure system, which takes suction from the SP, fails due to high pool temperature or excessive back pressure, and the low pressure systems are unavailable due to either random system faults or a failure to depressurize the RPV.

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 3.2.4.

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 Response to a Slow ATWS sequence

Figures 6.8 and 6.9 present the containment pressure and temperature time histories of a slow ATWS sequence calculated by the STCP code for Limerick (TC4 case of NUREG/CR-4624 [11]). In the calculation, the reactor is assumed to be at 30% rated thermal power. The containment pressure, as shown in Figure 6.8, reaches its failure point (130 psig) at about 40 minutes after accident initiation. The ECCS fails after containment failure, and core melt starts at about one hour. Vessel breach occurs at 150 minutes after accident initiation. Since a catastrophic containment failure (a failure area of 7 ft^2) is assumed in the calculation, containment pressure drops to atmospheric level and remains there after the containment fails. Containment temperature, as shown in Figure 6.9, reaches its design value early in the accident, drops slightly after containment failure, and then rises to about 600°F a few hours after vessel breach. It drops suddenly at about eight hours after accident initiation, when the drywell floor is penetrated by corium attack, and continues to decrease afterward because the corium has relocated to the suppression pool.

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 6.4 shows the challenges and the strategies and SAM actions required for a slow ATWS sequence. Results of the TC4 sequence, shown in Figures 6.8 and 6.9, are used to provide timing and containment conditions for Table 6.4. Table 6.4 is used to guide the discussions that follow.

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 a slow 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 6.4, the EPG control variables would exceed their limits within about 30 minutes after accident initiation because of the enormous 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. Although the RPV was assumed to be at high pressure in the calculation of Figures 6.8 and 6.9, RPV depressurization can be successful and the low pressure ECCS system is then available for core cooling if needed. Drywell cooling and drywell spray (if operated with RHR heat exchangers) would remove some energy from the containment, but their combined capacity (designed for about 2% of rated thermal power, Section 3.2.1) may 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 6.10 shows the effective venting area required to keep the containment pressure at a constant value for various energy input rates. (The curves in Figure 6.10 are based on isentropic 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 6.10 shows that an effective vent area of about 1.5 ft^2 is required to maintain containment pressure below 70 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) is reached in approximately 30 minutes 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. Because of the high reactor power, the vent area required to maintain the containment pressure at (or below) PCPL will be relatively large. The total vent area available for Limerick, i.e., the area discussed in Section 3.2.4, seems to be sufficient to maintain the containment pressure below the failure pressure if the reactor power can be reasonably controlled (e.g., less than 15% rated power). However, as discussed above, the real flow capacity of the vent lines needs to be assessed. Since containment venting is needed before core melt begins and the radioactivity in the containment atmosphere is low, actual venting could be started earlier, before PCPL is reached. The use of a hardened vent path eliminates the concern of potential damage to the RB structures and the equipment (Section 3.2.4.2).

In the ATWS sequences, the SP is saturated about 30 minutes after accident initiation. The opening of the vent paths will cause the containment to depressurize and the SP to flash. Consequently, in some Mark II plants, the pumps that take suction from the SP could fail. These pumps should be switched to an alternate suction source before the opening of any vent path (Section 3.2.1).

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 CAC system in the containment vent and purge mode (Sections 3.2.1) can be used to supply nitrogen to the containment after the reactor power is under control and the containment is depressurized.

6.3.2.2.2 Challenges and Strategies in the Late and Release Phases

Without successful containment venting, containment pressure would continue to rise until the containment fails. This will be followed, in some Mark II plants, by the loss of the ECCS pumps, core melt, and vessel breach. Before containment failure pressure is reached, the RPV may repressurize because the pneumatic supply pressure may not be sufficient to keep the SRVs open with the high containment back pressure. However, the decrease of containment pressure after containment failure would permit the RPV to depressurize again, and core melt and vessel breach would take place at a low RPV pressure. Since the containment has already failed in the very early phase of the accident, the primary objectives of CRM in these 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 Response to a Fast ATWS Sequence

Figures 6.11 and 6.12 present the containment pressure and temperature time histories, respectively, of a fast ATWS sequence for Limerick (TC3 case of NUREG/CR-4624 [11]). The difference between this case and the slow ATWS case is that in the present case core injection is lost, and core melt starts, before containment failure. The high pressure injection systems are lost due to either high SP temperature or high turbine exhaust back pressure, and the low pressure injection system is not available due to RPV depressurization failure.

Before the loss of core injection, the reactor power is high and the containment pressure rises rapidly. 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. Containment pressure stays almost level after the start of core melt because of the combined effects of lower energy input rate and energy absorption by containment heat structures. This leveling of containment pressure lasts until the core collapses (at about 70 minutes in Figure 6.11), after which the containment pressure increases considerably (about 30 psi, Figure 6.11). The high pressure vessel blowdown that follows results in a significant pressure rise in the containment (about 70 psi, Figure 6.11), and containment failure pressure is reached about one hour after VB. Containment temperature basically follows the same trend as containment pressure before containment failure, drops sharply after containment failure, and increases steadily during CCI to over 500°F. It drops sharply again at about six hours after accident initiation when the drywell floor is penetrated by corium attack, and continues to decrease afterward, when the corium is relocated to the suppression pool.

6.3.3.2 Challenges and Strategies of Fast ATWS Sequences

Table 6.5 shows the challenges as well as the strategies and SAM actions for a fast ATWS sequence. Results of the TC3 sequence, shown in Figures 6.11 and 6.12, are used to provide timing and containment conditions for Table 6.5. 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 are available for the ATWS sequence.

Since PCPL is not reached before vessel breach, containment venting is not likely to be carried out as a result of the EPGs. As shown in Figure 6.11, the venting pressure is reached during vessel blowdown. Since containment failure pressure is reached approximately one hour after VB, there would be sufficient time to open the vent path to prevent a containment overpressure failure. The situation after VB is similar to the fast SBO sequence discussed in Section 6.3. It should be noted again, however, that there are considerable uncertainties in both the containment pressure capability and the ability to accurately predict accident progression. The containment failure pressure could be reached during vessel blowdown, immediately after VB. Should this happen, there would not be sufficient time for containment venting, and early containment venting to reduce the base pressure before VB becomes more important.

Table 6.5 shows the challenges and strategies for this case. They are in general similar to those of Table 6.2 for the fast SBO sequences. However, the availability of electric power would improve the chances for implementation of some strategies. For example, containment cooling or containment spray may be available, and their use will remove energy from the containment atmosphere to reduce the base pressure and temperature. However, the effect may not be significant because the temperature of the containment atmosphere before vessel breach is not significant.

6.4 Loss of Containment Heat Removal Sequences

Accidents involving the loss of long term containment heat removal (CHR) 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 response required to mitigate the various challenges of a loss of CHR sequence is similar to that for a slow ATWS sequence (Table 6.4), but with much longer time windows available for action. The most important operator action, containment venting, is not required until more than about 20 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 BWR plants 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 application of CRM measures will be similar to that of a slow ATWS sequence discussed in Section 6.3.2.

6.5 Containment Bypass Sequences

Containment bypass 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 consequence sequence and will be discussed here.

There are 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 scenario 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 wetwell venting. The flow to the secondary containment from wetwell 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 Both AC & DC Power**	(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, DW/P, and DW/T (Note 1) *SP Cooling (Note 2)*
0:15	SP/T > 110 F	Potential Unacceptable SP Boundary Load	Enter EPG RPV Control G/L (Note 3)
0:30		(TSC Operational)	
	DW/P > 2 psig DW/T > 135 F	(EPG Entry Condition)	*CAC System and SGTS* *DW Cooling & WW Spray*
		Early Phase of the Accident	
2:00	**Core Melt Starts** *SP/T > HCTL* *DW/T> 340 F*	 Same as Above Containment Temperature Load	Early Recovery Actions Emergency RPV Depressurization *DW Spray* *RPV Depresurization*
	DW/P > SCSIP *DW/P Increase & VB Imminent*	Same as Above Potential Challenge at VB	*Early WW Venting* *Reactor Cavity Flooding* *DW Spray*
4:00	**Vessel Breach**		
		Load Associated with HPME (DCH, M&E Addition & SP Load)	*RPV Dep. or Above Actions Before VB to be Effective*

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)
4:00			**Late Recovery Actions**
	DW/P > PSP	Containment Pressure Load	RPV Depressurization
	CCI	Containment Pressure, Drywell Temperature, Noncondensible Gas Generation, DW Floor Melt-through, and SP Bypass	Corium Flooding Containment Flooding Containment Spray Containment Venting
10:00	**Containment Failure**		
		Release Phase of the Accident	
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		Corium Flooding Water to RPV & DW Cooling SP Cooling DW Spray and Containment Flooding
10:00 -	Containment Failure or Venting	FP Release from Containment to Outside	WW Venting Flooding Leak Area Containment Flooding
	FP & Pressure in SC	FP Release from SC to Environment	SGTS Fire Spray

Note:

1. SP/T - Suppression Pool (SP) Temperature, SP/L - SP Level, DW/P - Drywell or Containment Pressure, DW/T - Drywell 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 after battery depletion is very uncertain. Unless special arrangements have been made, they are generally not available. However, recovery of electic power 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, DW/P, and DW/T (Note 1) *SP Cooling (Note 2)*
0:15			Declare Site Area Emergency Activate EOF
0:30		(TSC Operational)	
	DW/P > 2 psig *DW/T > 135 F*	(EPG Entry Condition)	*CAC System and SGTS* *DW Cooling & WW Spray*
	SP/T > 110 F	Potential Unacceptable SP Boundary Load	Enter EPG RPV Control G/L (Note 3)
4:00	SP/T > HCTL	SP Boundary Load	Emergency RPV Depressurization
6:00	Battery Depletion	(Partial Recovery Will Change the Sequence to a TW Seq.)	General Emergency Declared Now or Earlier
		<u>Early Phase of the Accident</u>	
12:00	**Core Melt Starts**		Early Recovery Actions
	RPV Depressurization	SP Boundary Load	*Alternate E/P & P/S (Note 4)*
	DW/T > 340 F	Containment Temperature Load	*DW Spray* *RPV Depressurization*
	DW/P > SCSIP	SP Boundary Load	*DW Spray*
	DW/P Increase & VB Imminent	Potential Challenge at VB	*DW Spray, Early WW Venting* *Reactor Cavity Flooding*
16:00	**Vessel Breach**		
		Load Associated with HPME (DCH, M&E Addition & SP Load)	*RPV Dep. or Above Actions Before VB to be Effective*

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

Time Hr:Min	Plant Status	Challenge (Comment)	Strategy or SAM Actions
		Late Phase of the Accident	
			(Note 5)
16:00			Late Recovery Actions
	DW/P > PSP	Containment Pressure Load	RPV Depressurization
	CCI	Containment Pressure, Drywell Temperature, Noncondensible Gas Generations, DW Floor Melt-through, and SP Bypass	Cont. Cooling, DW Spray Corium Flooding & Containment Flooding
	DW/P > PCPL	Containment Pressure Load	DW Spray Containment Venting
21:00	**Containment Failure**		
		Release Phase of the Accident	
12:00 - 16:00	In-Vessel Release and Release at VB	FP in Containment Atmosphere	RPV Depressurization
16:00	Ex-Vessel Release (CCI) Revolatilization Late Iodine Release All Above		Corium Flooding Water to RPV & DW Cooling SP Cooling DW Spray and Containment Flooding
21:00	Containment Failure or Venting	FP Release from Containment to Outside	WW Venting Flooding Leak Area Containment Flooding
	FP & Pressure in SC	FP Release from SC to Environment	SGTS Fire Spray

Note:

1. SP/T - Suppression Pool (SP) Temperature, SP/L - SP Level, DW/P - Drywell or Containment Pressure, DW/T - Drywell 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 RPV will repressurize after the loss of SRV control power (i.e., dc power). RPV depressurization can be maintained 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 Consequences of the Various APBs in Figure 6.7

	Base Case	VT1	VT2	SV1	SV2
Latent Fatalities	3,040	209	139	100	91
50-Mile Population Dose (Person-sieverts)	1.5(5)*	1.8(4)	1.2(4)	7.2(3)	7.0(3)
Offsite Cost ($)	5.3(9)	4.8(7)	3.4(7)	1.2(7)	1.2(7)
Decontaminated Farm Area (Hectares)	70	0	0	0	0
Interdicted Farm Area (Hectares)	70,500	1,063	701	264	312

Note: *1.5(5) = 1.5×10^5

Table 6.4 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
		<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, DW/P, and DW/T (Note 1) SP Cooling
	SP/T > 110 F	Potential Unacceptable SP Boundary Load	Enter EPG RPV Control G/L (Note 2)
	DW/P > 2 psig	(EPG Entry Condition)	CAC System and SGTS DW Cooling & WW Spray
	SP/T > HCTL	SP Boundary Load	Emergency RPV Depressurization
	DW/T > 135 F	(EPG Entry Condition)	DW Cooling
	DW/P > SCSIP	SP Boundary Load	DW Spray
	DW/P > PSP	Containment Pressure Load	RPV Depressurization
	DW/T > 340 F	Containment Temperature Load	DW Spray RPV Depressurization
0:30	DW/P > PCPL	Containment Pressure Load	DW Spray Containment Venting
		(TSC Operational)	
0:40	**Containment Failure**	Release of Containment Atmosphere	(See Release Phase)

Table 6.4 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		
1:00	Core Melt Starts		Early Recovery Actions
2:30	Vessel Breach		
	Late Phase of the Accident		
2:30	CCI	Noncondensible Gas Generation, DW Floor Melt-through, and SP Bypass	Late Recovery Actions Corium Flooding Containment Flooding
	Release Phase of the Accident		
0:40 - 1:00	Containment Failure	Release of Containment Atmosphere	RPV Depressurization
1:00 - 2:30	In-Vessel Release and Release at VB	FP in Containment Atmosphere	RPV Depressurization
2:30 -	Ex-Vessel Release (CCI) Revolatilization Late Iodine Release All Above		Corium Flooding Water to RPV & DW Cooling SP Cooling DW Spray and Containment Flooding
1:00 -	Containment Failure or Venting	FP Release from Containment to Outside	WW Venting Flooding Leak Area Containment Flooding
	FP & Pressure in RB	FP Release from RB to Environment	SGTS Fire Spray

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

Table 6.5 Challenges and Strategies for a Fast 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
		<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, DW/P, and DW/T (Note 1) SP Cooling
	SP/T > 110 F	Potential Unacceptable SP Boundary Load	Enter EPG RPV Control G/L (Note 2)
	DW/P > 2 psig DW/T > 135 F	(EPG Entry Condition)	CAC System and SGTS DW Cooling & WW Spray
	SP/T > HCTL	SP Boundary Load	Emergency RPV Depressurization
0:30		(TSC Operational)	
		<u>Early Phase of the Accident</u>	
0:35	**Core Melt Starts** DW/P > SCSIP	SP Boundary Load	Early Recovery Actions DW Spray
	DW/P > PSP	Containment Pressure Load	RPV Depressurization
	High Presure VB Imminent	Potential Challenge at VB	Early WW Venting Reactor Cavity Flooding DW Spray
1:40	**Vessel Breach**		
		Load Associated with HPME (DCH, M&E Addition & SP Load)	RPV Depressurization or Above Actions Before VB to be Effective

Table 6.5 Challenges and Strategies for a Fast [illegible] nce ([illegible] nued)

Time Hr:Min	Plant Status	Challeng [illegible]	Strategy or SAM Actions
		Late Phase of the Accide	
1:40			Late Recovery Actions
	DW/T > 340 F	Containment Temperature Load	DW Spray RPV Depressurization
	DW/P > PCPL	Containment Pressure Load	DW Spray Containment Venting
	CCI	Containment Pressure, Drywell Temperature, Noncondensible Gas Generation, Drywell Floor Melt-Through, and Suppression Pool Bypass	Corium Flooding Containment Flooding
2:40	**Containment Failure**		
		Release Phase of the Accident	
0:35 - 1:40	In-Vessel Release and Release at VB	FP in Containment Atmosphere	RPV Depressurization
1:40 -	Ex-Vessel Release (CCI) Revolatilization Late Iodine Release All Above		Corium Flooding Water to RPV & DW Cooling SP Cooling DW Spray and Containment Flooding
2:40 -	Containment Failure or Venting	FP Release from Containment to Outside	WW Venting Flooding Leak Area Containment Flooding
	FP & Pressure in RB	FP Release from RB to Environment	SGTS Fire Spray

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

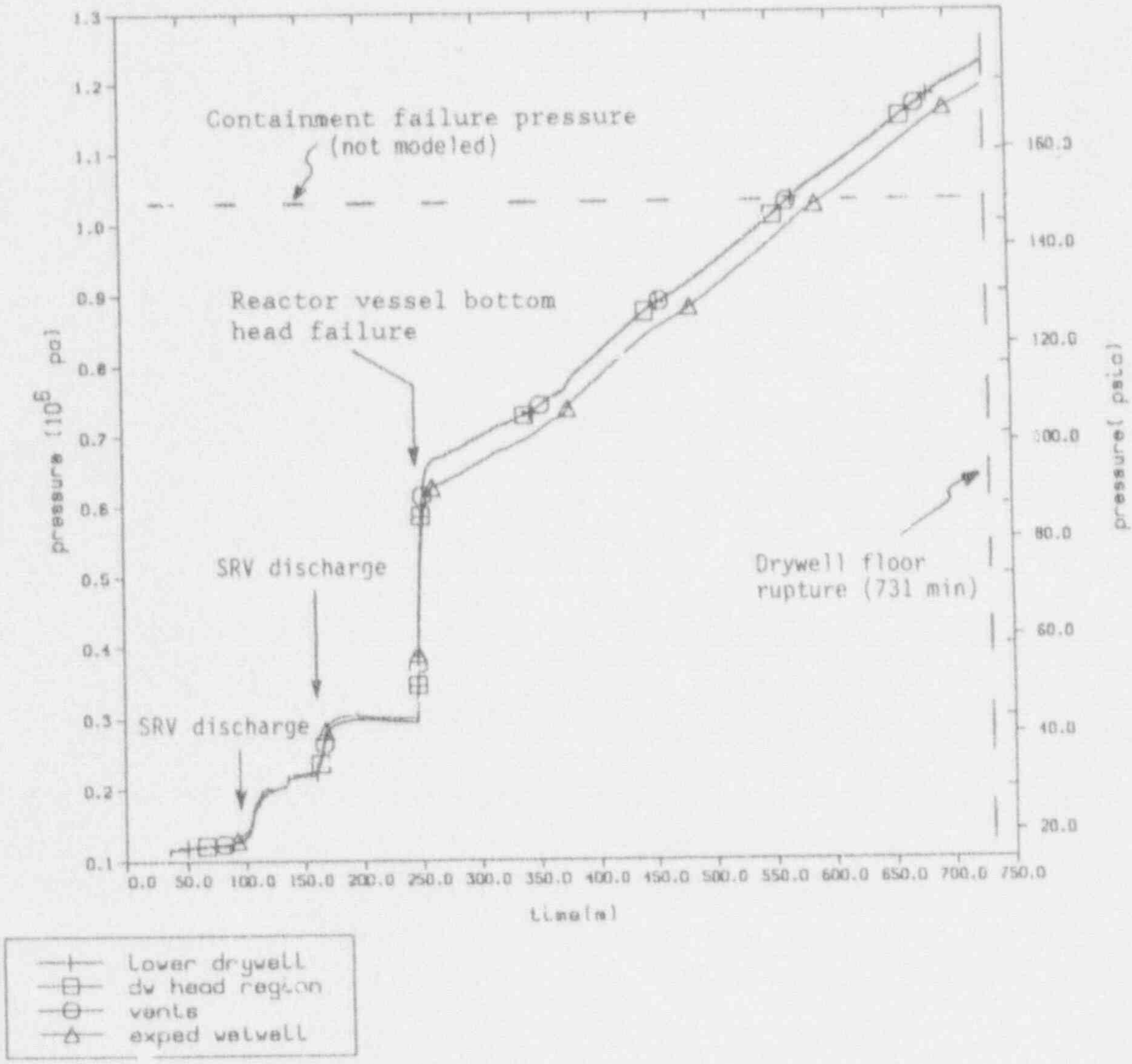

Figure 6.1 Primary Containment Pressure Distribution for the Mark II Short-Term Station Blackout Sequence Without ADS Actuation (Reference 13)

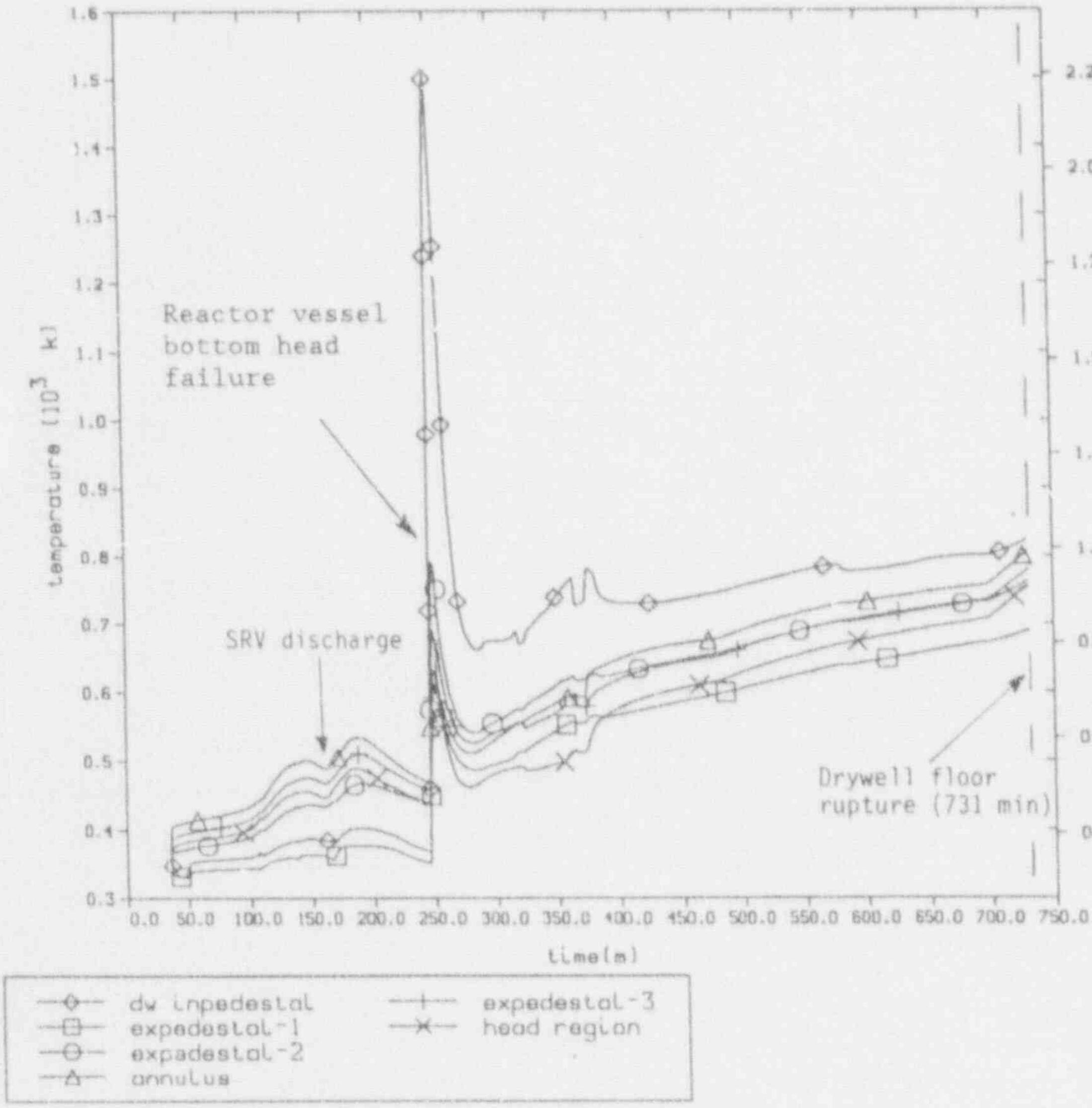

Figure 6.2 Primary Containment Temperature Distribution for the Mark II Short-Term Station Blackout Sequence Without ADS Actuation (Reference 13)

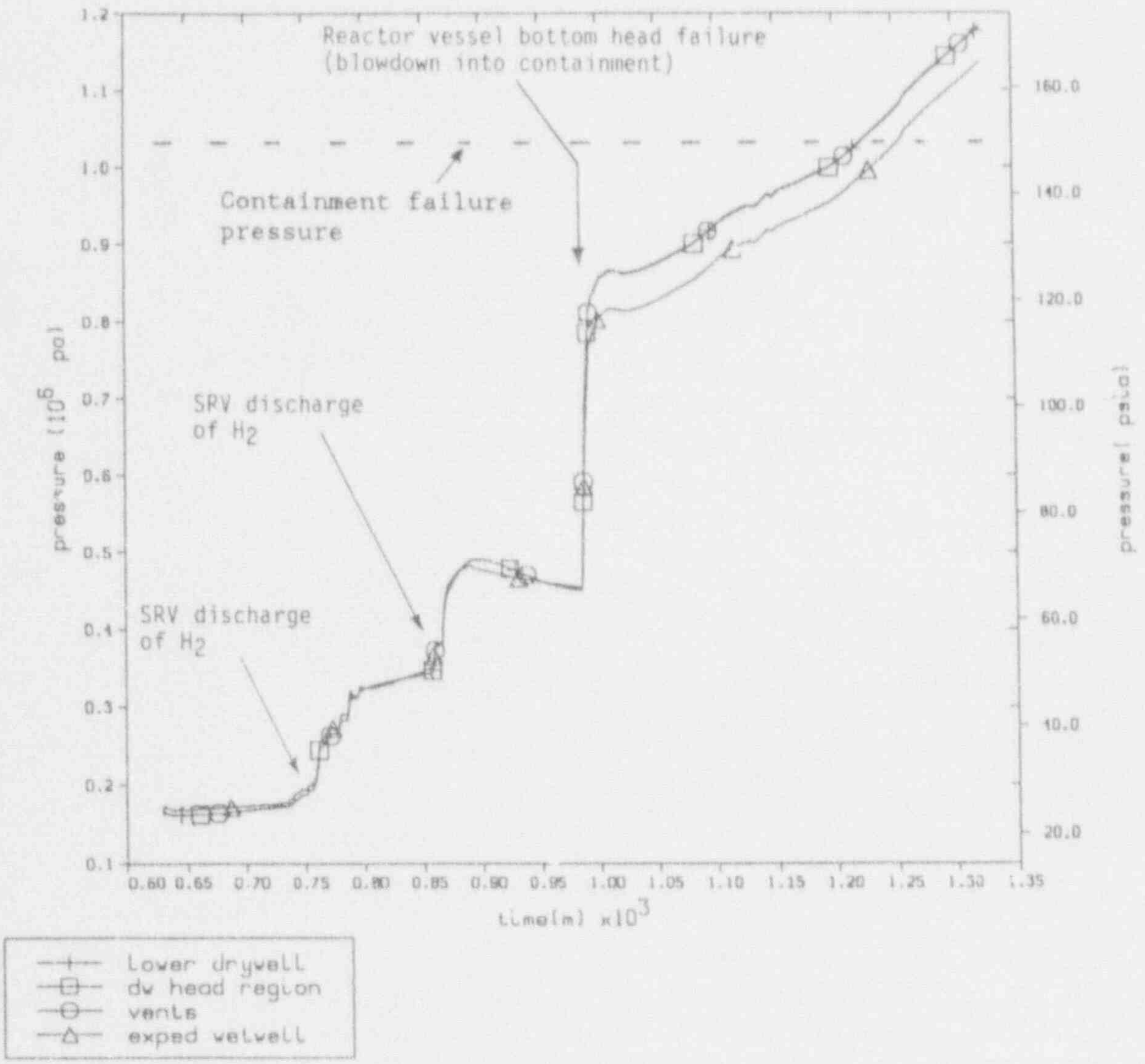

Figure 6.3 Primary Containment Pressure Distribution for the Mark II Long-Term Station Blackout Sequence With ADS Actuation (Reference 13)

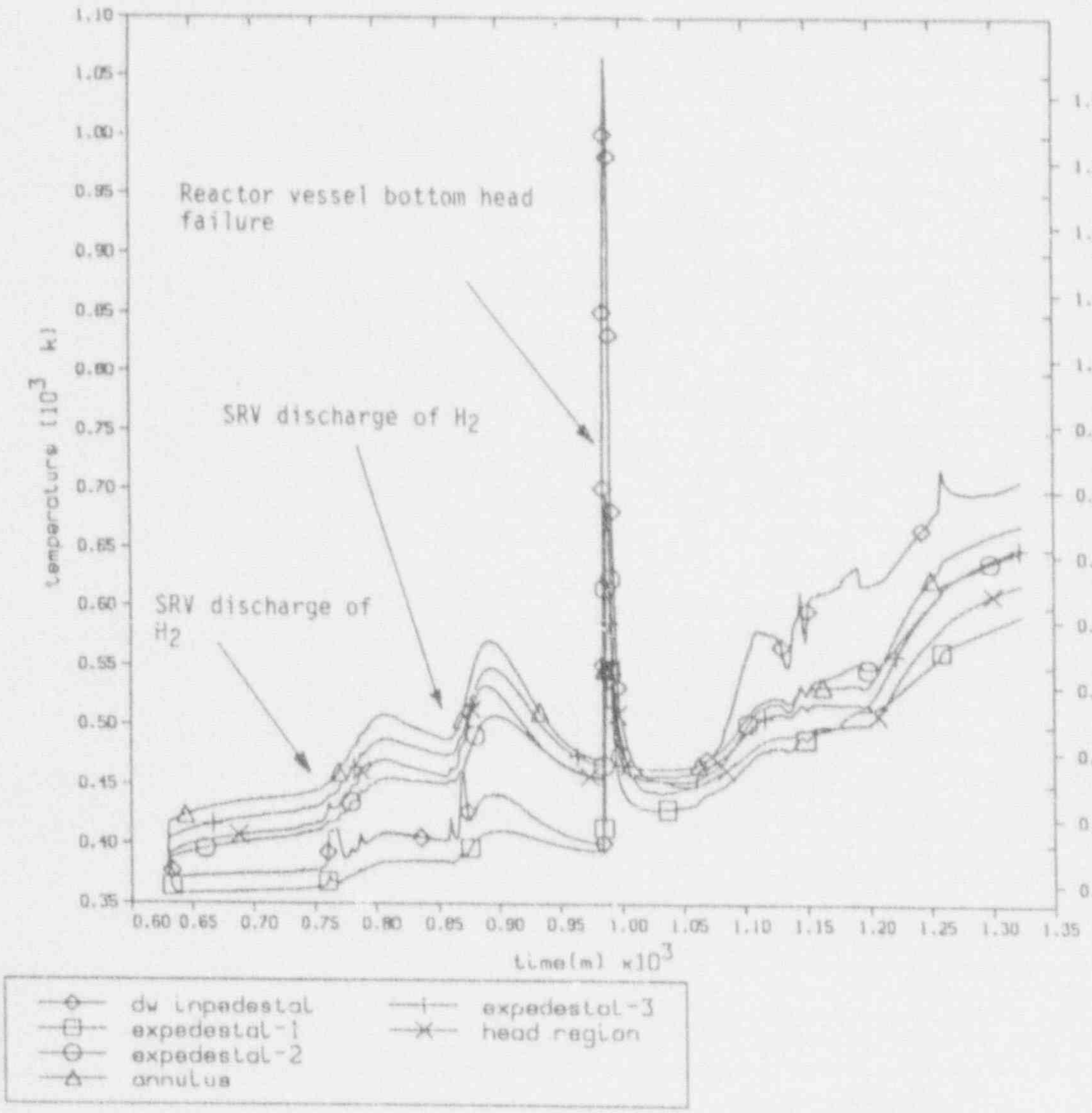

Figure 6.4 Primary Containment Temperature Distribution for the Mark II Long-Term Station Blackout Sequence (Reference 13)

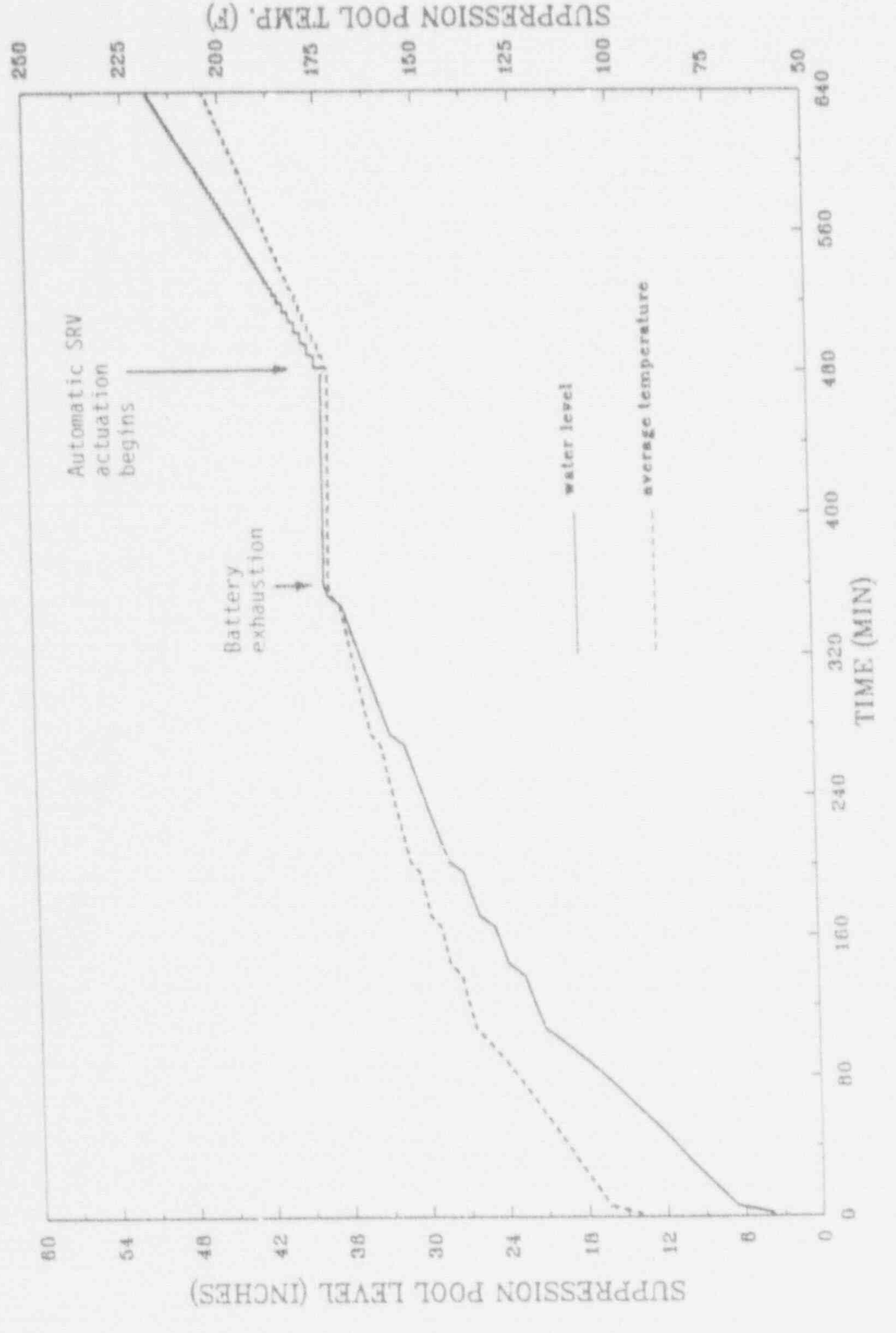

Figure 6.5 Pressure Suppression Pool Water Level and Average Temperature for the Mark II Long-Term Station Blackout Sequence (Reference 13)

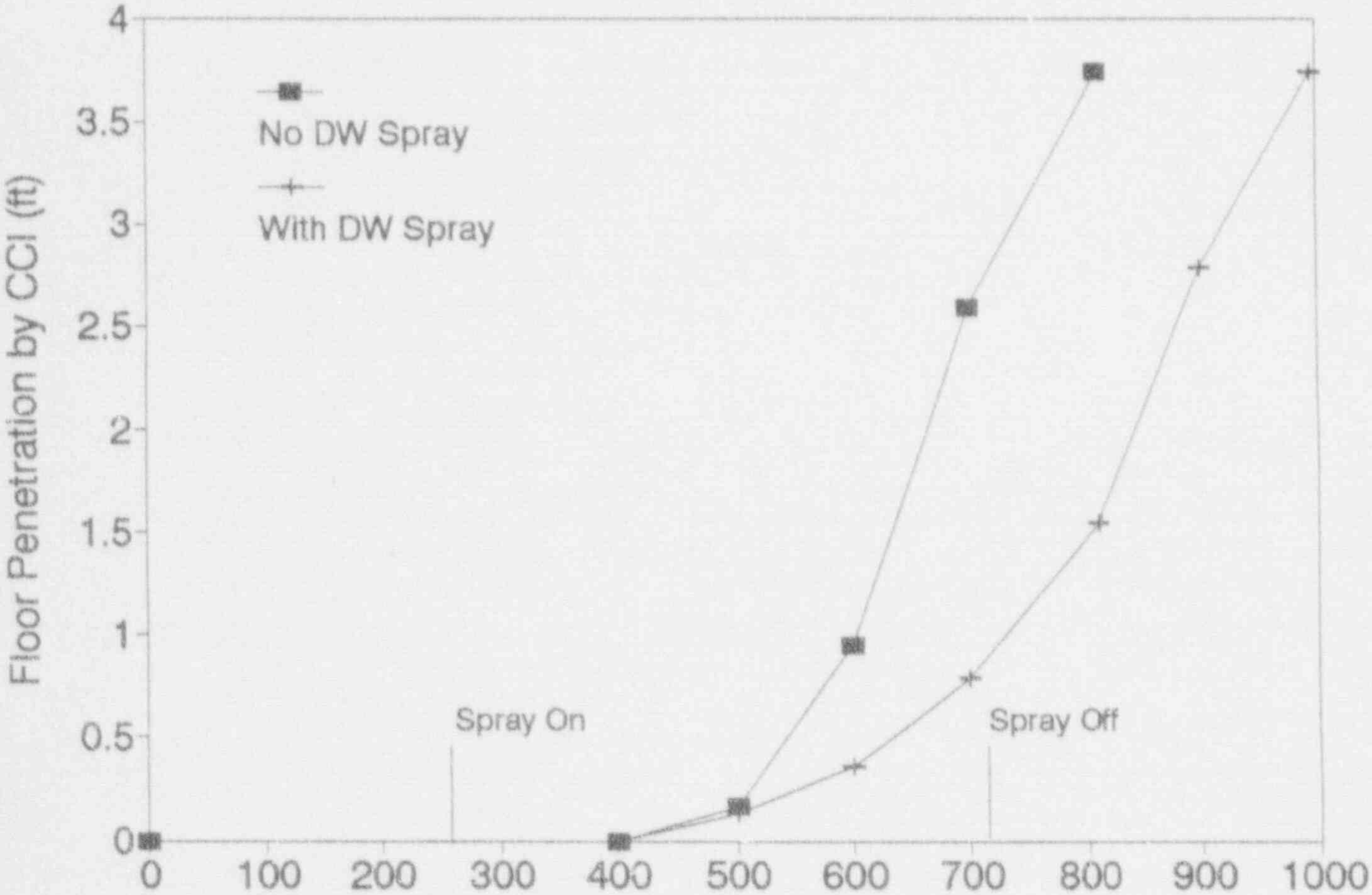

Figure 6.6 Reactor Cavity Floor Penetration by CCI for Fast SBO Sequences With RPV Depressurization (Data From Reference 13)

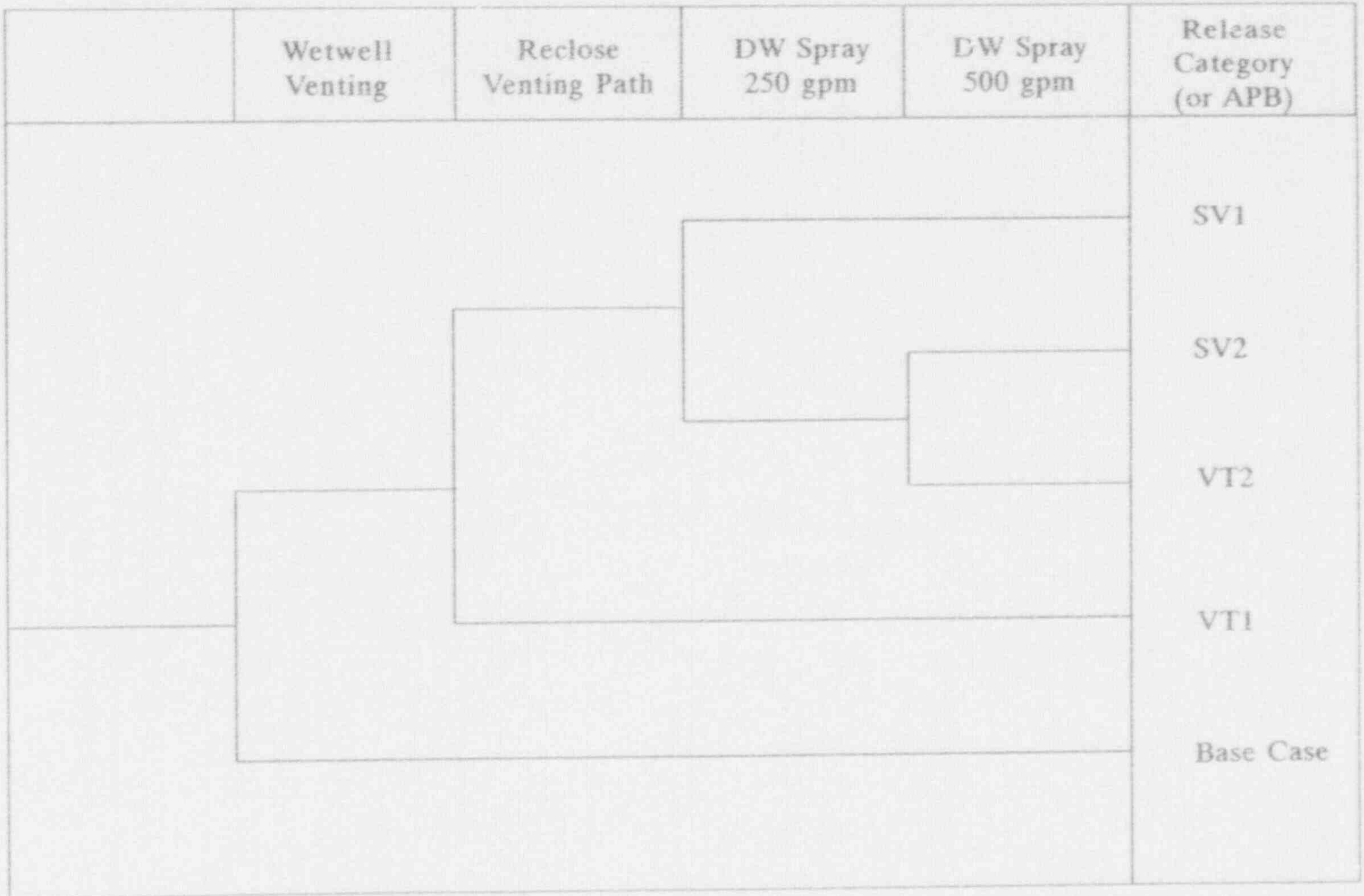

Figure 6.7 Partial Release CRET for FP Release Comparison

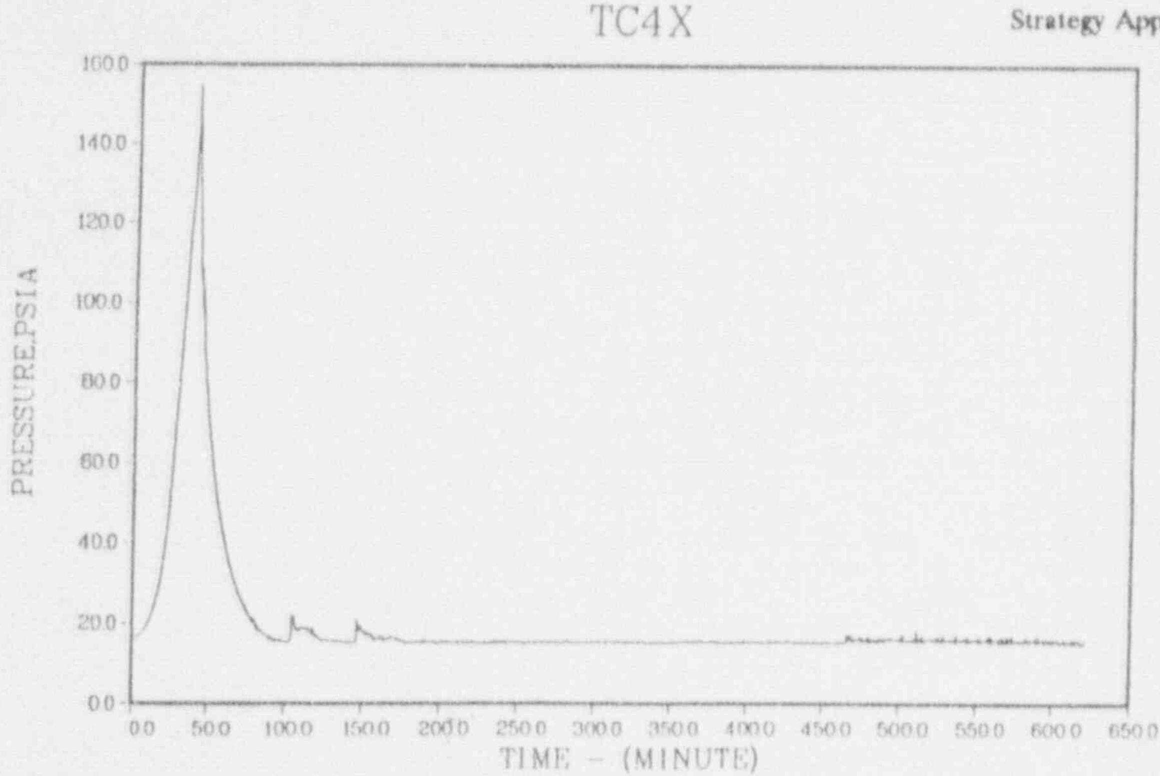

Figure 6.8 Containment Pressure Response for Limerick TC4 (slow ATWS) Sequence

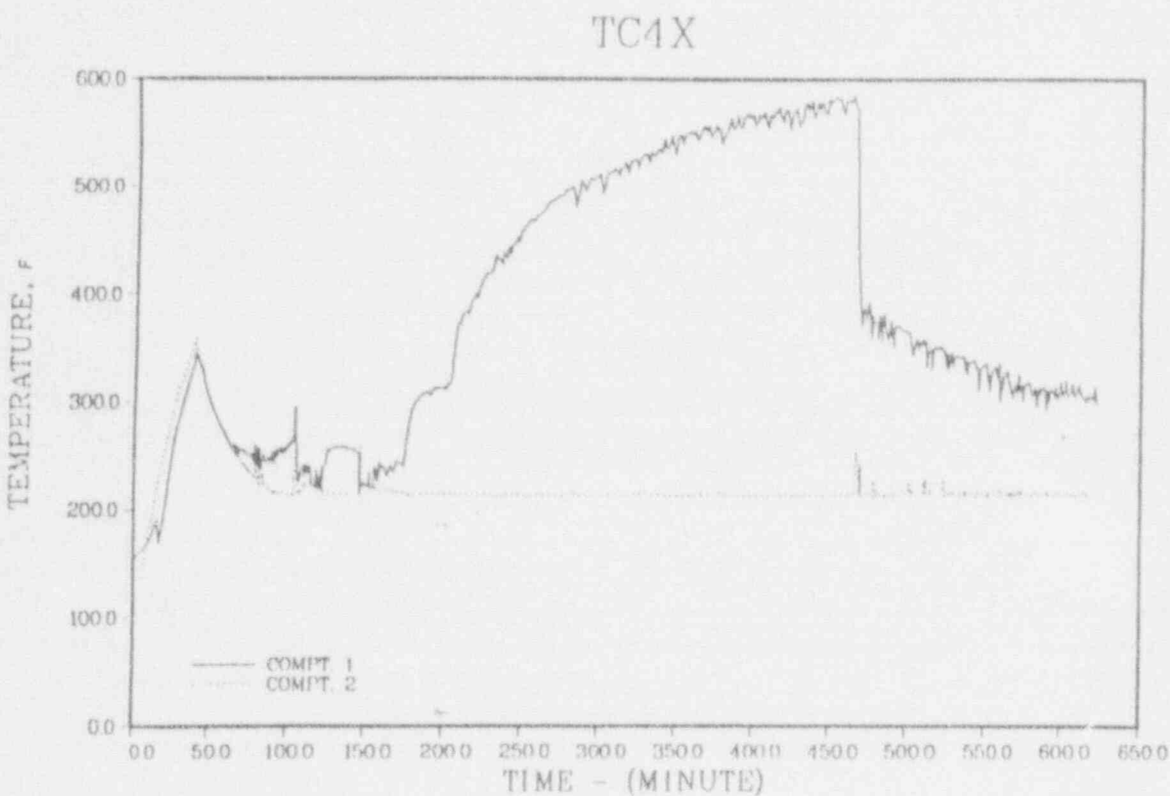

Figure 6.9 Containment Temperature Response for Limerick TC4 (slow ATWS) Sequence

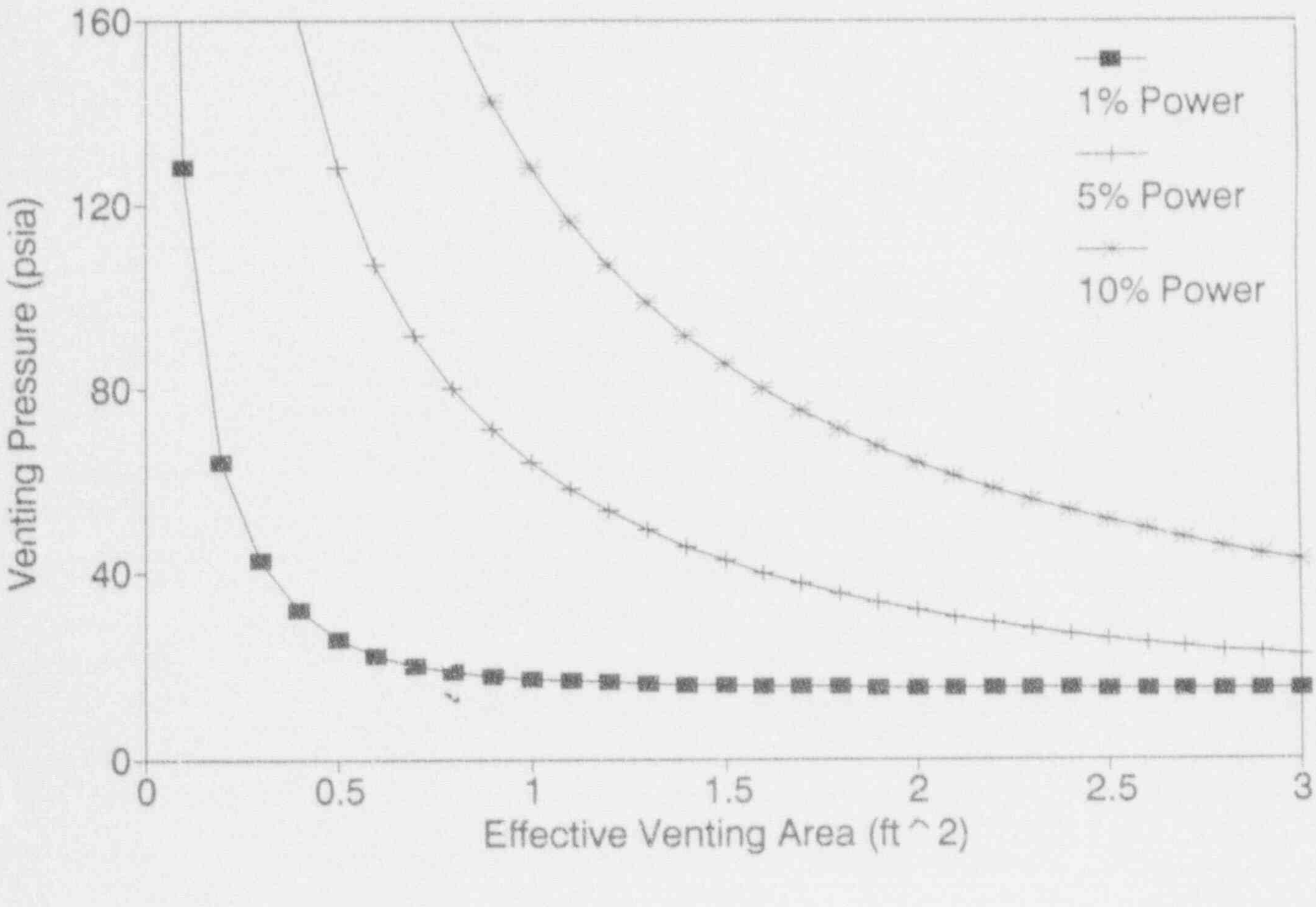

Figure 6.10 Venting Area and Venting Pressure for Various Reactor Power Levels

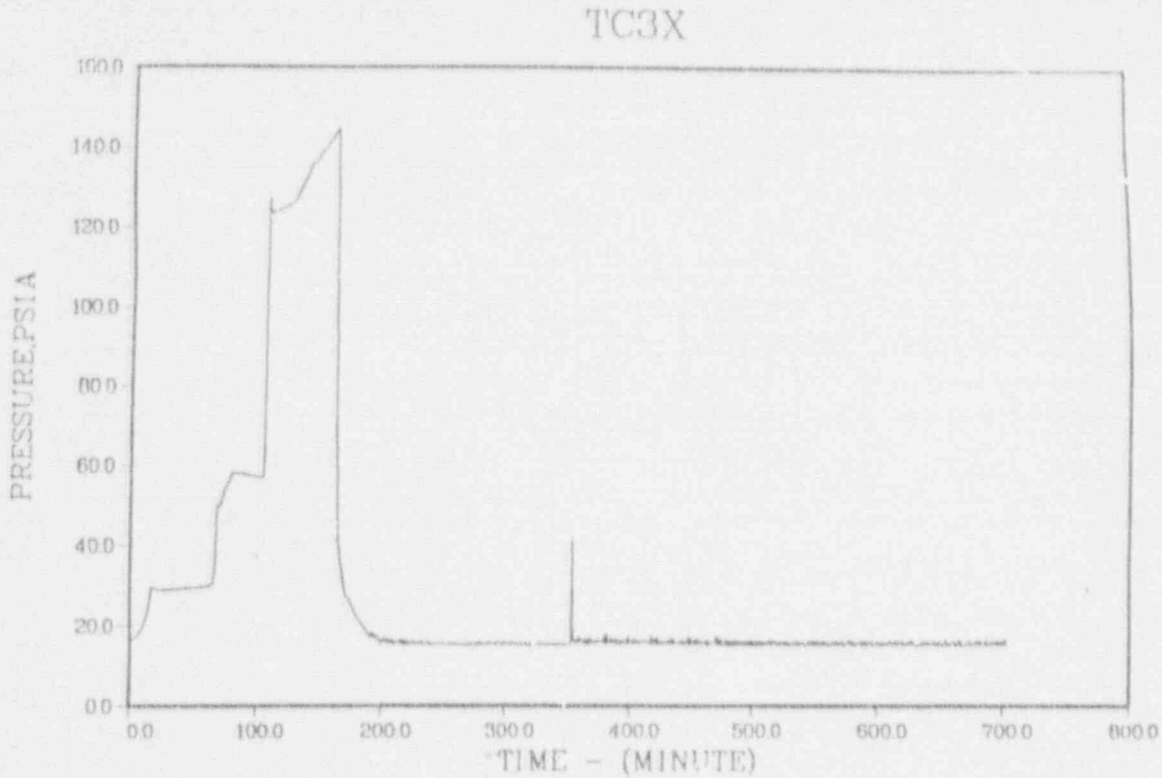

Figure 6.11 Containment Pressure Response for Limerick TC3 (Fast ATWS) Sequence

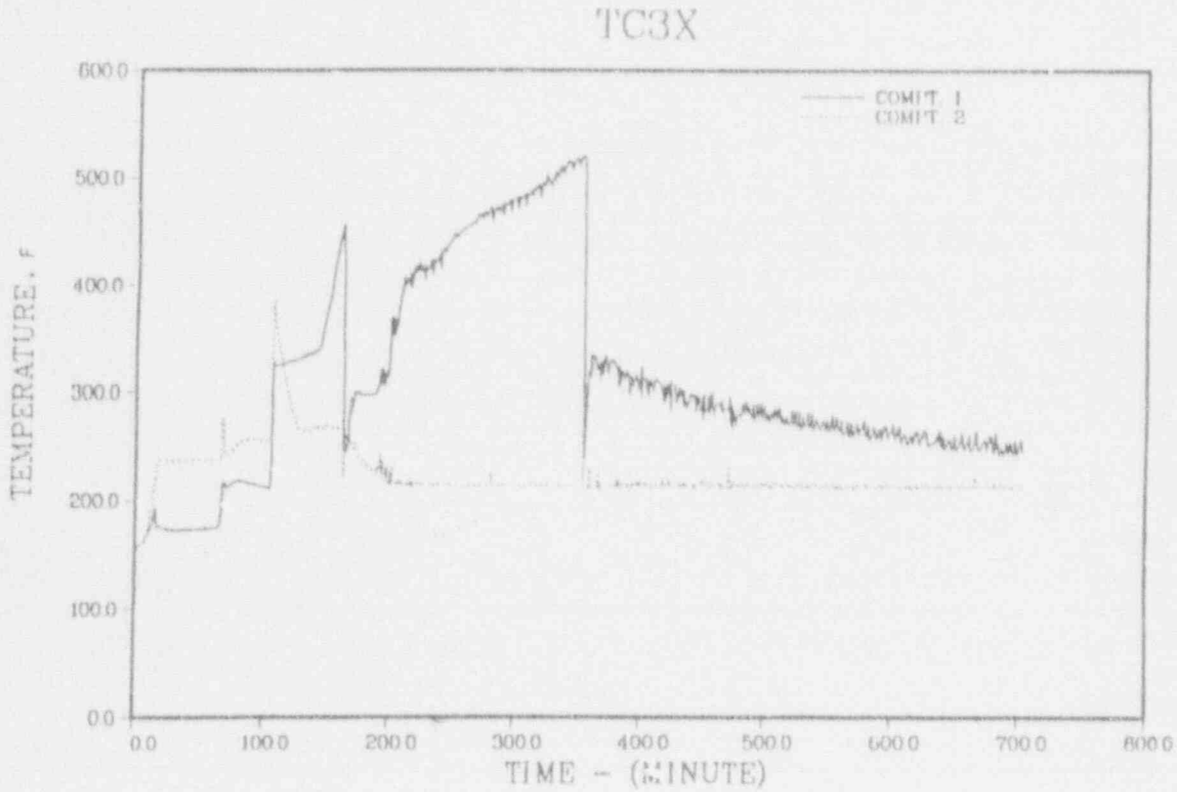

Figure 6.12 Containment Temperature Response for Limerick TC3 (Fast ATWS) Sequence

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 II 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 II 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.

Existing Instrumentation and Environmental Qualification: The most significant potential problem with plant instrumentation for CRM is the lack of sufficient control room indications of containment variables during a station blackout (SBO) sequence. According to NUREG-1150 and other PRAs, SBO is one of the most important severe accident sequences for Mark II containments. This lack of indicators severely restricts the ability of plant personnel to perform CRM activities during an SBO. To improve severe accident management, an alternate electric power supply beyond that required by existing regulations has been recommended by the CPI program for both RPV depressurization and containment venting. It may be desirable to provide such an alternate power supply also for some instrumentation that is important for making decisions on CRM activities. In addition, the identification of alternate methods to obtain containment variable indications in the absence of electric power will improve the availability of relevant information for CRM.

The survival of plant instruments under severe accident conditions could also be a problem. The containment conditions, e.g. temperature, pressure, and radiation, that may occur in a severe accident may exceed the environmental conditions for which the equipment and instruments are qualified. Even though the equipment and instruments may survive under conditions well beyond their qualification conditions, their accuracy is not assured.

A case by case analysis of the various types of instruments may be needed to determine their availability and reliability under severe accident conditions.

In some cases, the range of existing instruments does not cover the values the measured variables can assume in a severe accident. Examples include the drywell temperature reading (typical present range: 40 to 440°F) and the suppression pool water level indication (5 ft above normal water level). Presently there is no water level gauge in the drywell. This would be helpful if containment flooding is considered for containment and release management. For some challenges, e.g., ex-vessel steam explosion, there is no direct instrument indication and the identification of the potential for the challenge must rely on indirect indications.

7.2 Interface Between Existing EPGs 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.

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

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 stage 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 R[illegible] 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 however, it may be more 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.

However, the limiting conditions in the current EPGs regarding some emergency actions may not apply in the severe accident regime. An example is the exclusion plot for containment spray. This plot in the BWR EPGs prohibits the use of sprays under some temperature and pressure conditions in the containment because of the concern that the spray will quickly cool down the atmosphere and create a vacuum that may lead to a loss of containment integrity. The generic calculations for this plot assume a fixed content of non-condensible gases and a fixed temperature of the spray water. Both of these variables can change dramatically during the course of a severe accident. Hydrogen, carbon monoxide, and carbon dioxide will increase the non-condensible gases, and heat-up of the suppression pool will increase spray water temperature. Therefore, the use of sprays may indeed be beneficial in the later stages of a severe accident as discussed in previous sections, e.g., FP scrubbing, even though they may be prohibited under present EPG entry conditions.

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 secondary containment 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 prescribed procedure may be inappropriate for the particular situation. Some strategies maybe 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 containment and release management. (More detailed discussion on this issue can be found in the Mark I report [7].)

7.3 CRM Strategies

The CRM strategies in this report 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 4.2. 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.

Although there are significant uncertainties, the strategies listed in Table 5.2 were found to be in general effective based on their application during accident sequences calculated for the Limerick plant. The strategies should be beneficial for other BWR plants with Mark II containments. The effectiveness of strategies could be evaluated by including them explicitly in probabilistic safety analyses.

The decision to carry out a strategy during a severe accident, particularly those strategies with significant adverse effects such as venting, 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 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 severe accident 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 quantify 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.

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11. ABSTRACT (200 words or less)

Accident management strategies that have the potential to maintain containment integrity and control or mitigate the release of radioactivity following a severe accident at a boiling water reactor with a Mark II type of containment are identified and evaluated. The strategies are referred to as containment and release strategies. Using information available from probabilistic risk assessments and other existing severe accident research, and employing simplified containment and release trees, this report identifies the challenges a Mark II containment may encounter during a severe accident, the mechanisms behind these challenges, and the strategies that could be used to mitigate the challenges. By means of a safety objective tree, the strategies are linked to the general safety objectives of containment and release management. As part of the assessment process, the strategies are applied to certain severe accident sequence categories deemed important to a Mark II containment. These sequence categories exhibit 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. The Limerick Generating Station is used as a representative Mark II plant to illustrate plant specifics in this report.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

BWR Type Reactors- Containment Systems, Fission Product Release, Management, Reactor Accidents, Probability, Core Damage, Loss of Coolant, Radiation Protection, Failure Mode Analysis, Risk Assessment, Evaluation, Limerick-1 Reactor, Limerick-2 Reactor

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