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

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Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

This report identifies and assesses accident management strategies which could be important for preventing containment failure and/or mitigating the release of fission products during a severe accident in a BWR plant with a Mark I type of containment. Based on information available from probabilistic risk assessments and other existing severe accident research, and using simplified containment and release event trees, the report identifies the challenges a Mark I containment could face during the course of a severe accident, the mechanisms behind these challenges, and the strategies that could be used to mitigate the challenges. A safety objective tree is developed which provides the connection between the safety objectives, the safety functions, the challenges, and the strategies. The strategies were assessed by applying them to certain severe accident sequence categories which have one or more of the following characteristics: have high probability of core damage or high consequences, lead to a number of challenges, and involve the failure of multiple systems.

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EXECUTIVE SUMMARY

The purpose of the present report is to identify, as well as to assess, accident management strategies which could be important for preventing containment failure and/or mitigating the release of fission products during a severe accident in a BWR plant with a Mark I 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 present report emphasizes the use of existing plant capabilities for severe accident management. The plant features that are important to containment and release management (CRM) of a BWR Mark I containment are reviewed to identify their function and performance under severe accident conditions. The plant features that are reviewed in this report include the plant systems, the resources needed to support the operation of these systems, the emergency response facilities, the emergency procedure guidelines (EPGs), and the instrumentation required to assess the plant and its environs during and following an accident. Important issues related to these systems and some of the uncertainties involved in severe accident phenomena are discussed.

In order to optimize the effort of strategy identification and assessment, maximum use was made of previously available information from Mark I related safety studies. Through the use of a simplified event tree structure (i.e., the containment and release event tree, CRET), the existing data base is systematically reviewed to identify the challenges a Mark I containment could face during the course of a severe accident, the mechanisms behind these challenges, and the strategies that can be used to mitigate these challenges. 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.

Important CRM strategies identified by the above processes are discussed in detail in this report to provide guidance for the development of symptom based strategies which could be considered for implementation. The most important points related to strategy implementation are discussed with emphasis on symptoms leading to strategy initiation, diagnostic concerns, downside risks, and operator action concerns.

The strategies were assessed by application to certain accident sequences. The sequence categories selected for strategy assessment consisted of station blackout, ATWS, LOCA, 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. 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.

The Peach Bottom Atomic Power Station (PBAPS) was used as the surrogate plant for this assessment. The PBAPS Transient Response Implementation Plan (TRIP), a plant specific implementation of the BWR Emergency Procedure Guidelines (EPGs), was used to determine the operation response as currently expected at the plant. While the existing EPGs are designed primarily for plant conditions expected prior to significant core damage, CRM strategies consider plant conditions well beyond this point, including vessel breach and containment failure where release management becomes more important. Existing data from Mark I related analyses were used in the strategy assessment. In cases where additional data were needed to address the effect of a particular strategy the Source Term Code Package (STCP) was used to provide an estimate of such an effect.

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 accident sequences calculated for the Peach Bottom plant. While in some cases 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 drywell atmosphere, and provide water for corium quenching), more than one strategies would be needed in some other cases to control accident progression. For example, containment venting can be used to prevent the containment from reaching the primary containment pressure limit (PCPL), but containment temperature will continue to rise and the containment could fail by a combination of pressure and temperature loads if hot gases are released to the containment atmosphere such as might happen during core concrete interaction (CCI). Drywell spray, or other strategies that can be used to remove energy from the drywell atmosphere, may be needed in addition to venting to prevent containment failure.

Some of the strategies may have significant adverse effects. For example, venting will often result in the release of fission products to the environment. The decision to carry out a particular strategy that has potential adverse effects is further

complicated by the uncertainties in the ability to predict accident progression accurately and the uncertainties in estimating plant capabilities such as containment strength.

During an accident decision making for accident management may be enhanced through the use of CRETs 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.

The lack of control room indications of containment variables could be a significant problem for accident management. This deficiency is particularly serious for a station blackout (SBO) sequence, a dominant severe accident sequence for a Mark I containment according to NUREG-1150. The survival of plant instruments under severe accident conditions is also 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. Additional research efforts are needed in these areas.

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LIST OF ACRONYMS

ADS	Automatic Depressurization System
ANS	American Nuclear Society
APB	Accident Progression Bin
APET	Accident Progression Event Tree
ARM	Area Radiation Monitors
ATWS	Anticipated Transient Without Scram
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CACS	Containment Atmosphere Control System
CADS	Containment Atmosphere Dilution System
CCI	Core Concrete Interaction
CDF	Core Damage Frequency
CET	Containment Event Tree
CF	Containment Failure
CHR	Containment Heat Removal
CPI	Containment Performance Improvement
CR	Contractor Report
CRD	Control Rod Drive
CRDHS	Control Rod Drive Hydraulic System
CRET	Containment and Release Event Tree
CRM	Containment and Release Management
CST	Condensate Storage Tank
DCH	Direct Containment Heating
DF	Decontamination Factor
DSIL	Drywell Spray Initiation Limit
DSM	Drywell Shell Melt-Through
DW	Drywell
ECCS	Emergency Core Cooling System
EHS	Emergency Heat Sink
EOF	Emergency Operation Facility
EOP	Emergency Operating Procedures
EPG	Emergency Procedure Guidelines
FCCI	Fraction of Total FP Inventory Released During CCI
FLI	Fraction of Total Iodine Inventory from Late Release from Water Pools
FP	Fission Product
FRCS	Fraction of Total FP Inventory Released from RCS
FREV	Fraction of Total FP Inventory Released from Late Revolatilization
FW	Fire Water
HCLL	Heat Capacity Level Limit
HCTL	Heat Capacity Temperature Limit
HEPA	High Efficiency Particulate Air
HPCI	High Pressure Core Injection

LIST OF ACRONYMS (Cont'd)

HPME	High Pressure Melt Ejection
HPSW	High Pressure Service Water
HVAC	Heating Ventilating and Air Conditioning
IDCOR	Industry Degraded Core Rulemaking
IEEE	Institute of Electrical and Electronics Engineering
ILRT	Integrated Leak Rate Testing
INEL	Idaho National Engineering Laboratory
ISL	Interfacing Systems LOCA
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
LPCI	Low Pressure Core Injection
LPCS	Low Pressure Core Spray
MAAP	Modular Accident Analysis Program
MCDF	Mean Core Damage Frequency
MSIV	Main Steam Isolation Valve
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
OSC	Operational Support Center
PBAPS	Peach Bottom Atomic Power Station
PCPL	Primary Containment Pressure Limit
PDS	Plant Damage State
PIV	Pressure Isolation Valve
PRA	Probabilistic Risk Assessment
PSP	Pressure Suppression Pressure
PWR	Pressurized Water Reactor
RB	Reactor Building
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Cooling System
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RWST	Refueling Water Storage Tank
SARP	Severe Accident Research Plan
SARRP	Severe Accident Risk Reduction/Risk Rebaselining Program
SASA	Severe Accident Sequence Analysis
SBO	Station Blackout
SCSIP	Suppression Chamber Spray Initiation Pressure
SGIG	Safety Grade Instrument Gas
SGTS	Standby Gas Treatment System
SLC	Standing Liquid Control
SNL	Sandia National Laboratory
SOT	Safety Objective Tree
SORV	Stuck Open Relief Valve

LIST OF ACRONYMS (Cont'd)

SP	Suppression Pool
SPDS	Safety Parameter Display System
SRV	Safety Relief Valve
STCP	Source Term Code Package
TAF	Top of Active Fuel
TMCDF	Total Mean Core Damage Frequency
TMI	Three Mile Island
TPLL	Tail Pipe Level Limit
TRIP	Transient Response Implementation Plan (Peach Bottom)
TSC	Technical Support Center
VB	Vessel Breach

1. INTRODUCTION

1.1 Background

Experience obtained from Probabilistic Risk Assessment analyses indicates that a cost effective means for licensees to reduce severe accident risk even further is to supplement plant operating procedures with additional 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" [2], and NUREG/CR-5132, "Severe Accident Insights Report" [3]. In March 1990 NUREG/CR-5474, "Assessment of Candidate Accident Management Strategies" [4] 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 [5], were assessed to provide information to individual licensees for consideration when performing their Individual Plant Examinations. The assessment focused on describing and explaining the strategies, considering their relationship to existing requirements and practices, as well as identifying possible associated adverse effects. The emphasis of the strategies assessed in NUREG/CR-5474 was on preventing or mitigating core damage, i.e., on arresting the accident progression in-vessel. The effects of the strategies considered were generally well understood and many of the strategies were found to be already implemented at some plants.

The current phase of the NRC Research effort in identifying and assessing accident management actions is concerned with mitigative strategies which would most likely be applied in the more advanced stages of a severe accident [6,38]. Before vessel failure the emphasis is on arresting or mitigating core damage progression in the reactor vessel. If vessel failure has already occurred or is imminent the emphasis is on maintaining containment integrity, quenching core debris ex-vessel, and minimizing fission product release to the environment. While identification and assessment of advanced in-vessel strategies is being addressed by other NRC contractors, BNL is producing a series of reports dealing with the containment and release management part of a severe accident. The present report is one of this series. The mitigative strategies discussed here are often applied in situations where present understanding of the phenomena encountered

is limited. Therefore, the uncertainty for these strategies is larger than for the strategies examined in NUREG/CR-5474. Also, many of the suggested strategies go well beyond existing procedures. Often the strategies and the challenges which they address depend on the specific containment types and therefore five individual reports are being written for containment and release management, each one addressing the challenges and strategies applicable to one of the five containment types used in the U.S. today.

1.2 Objective and Scope

The purpose of the present report is to identify, as well as to assess, accident management strategies which could be important for preventing or delaying containment failure and/or mitigating the release of fission products during a severe accident in a BWR plant with a Mark I 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 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 program.

The report can also furnish the reviewer of an accident management plan with a systematic overview of the challenges a Mark I 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 I containment during a severe accident are discussed and strategies which can eliminate, or mitigate the effect of, most of these challenges are identified. Not all challenges can be completely 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 I 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 I plant are presented in Section 4. At the end of Section 4 the challenges and strategies are systematically arranged in a "Safety Objective Tree." Section 5 presents the pertinent information for each of the strategies in a consolidated form. The application of the strategies during certain accident sequences is discussed in Section 6. Section 7 consists of a summary and conclusions. References are contained in Section 8.

2. APPROACH TO STRATEGY IDENTIFICATION AND ASSESSMENT

2.1 General Information

In order to optimize the effort of strategy identification and assessment maximum use was made of previously available information from Mark I related safety studies. The sources used most frequently are the NUREG-1150 study and the supporting reports related to Mark I [7-9], the Mark I information produced for the NRC's SASA [10-14], SARPP [15-16] and CPI programs [17], and the relevant IDCOR reports [18-19].

The BWR Owners' Group's Emergency Procedure Guidelines (EPGs), Revision 4 [20], were used as a baseline to gauge the guidance presently available to BWR Mark I 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 I plants will currently respond to a severe accident.

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

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

In the discussions which follow, when it is instructive to refer to specific plant features, Peach Bottom Atomic Power Station is used as the example Mark I plant.

2.2 Strategy Identification Process

Numerous sources, referenced throughout this report, were consulted to obtain information on the challenges a Mark I 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 [4] 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 [21]. 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 I containment is illustrated in Figure 4.15.

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 I 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 to shortly after vessel breach, (3) the late phase, after vessel breach but prior to containment failure, and (4) a radiological release phase. These phases need not all occur in order. Depending on the accident, the radiological release phase can be entered from any of the other phases. Similarly, depending on the sequence and/or accident management actions, a recovery can be made from any of the first three phases. Figure 4.3 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 I 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 I Containment System

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

Enclosing the primary containment is the secondary containment, which includes the reactor building, the reactor building heating and ventilation system, and the standby gas treatment system (SGTS). The secondary containment is designed to provide a controlled, filtered and elevated release (through a stack) of the reactor building atmosphere.

There are 24 BWR facilities with Mark I containment system in the United States. Table 3.1 provides a listing of these units, and the operator and rated electric power of each unit. Figure 3.1 shows a schematic of the containment design for Peach Bottom. The drywell is a steel pressure vessel supported in concrete with a spherical lower section and a cylindrical upper section (light bulb shape). The suppression chamber is a steel pressure vessel in the shape of a torus located below the drywell and encircling it. These constructions are typical of all, but two, Mark I units. Instead of steel vessels, the two units (Brunswick 1 and 2) utilize a steel-lined, reinforced concrete drywell and suppression chamber.

Table 3.2 gives the values of some plant parameters for Peach Bottom and the ranges of these parameters for all domestic Mark I plants. As shown in Table 3.1, the rated thermal power varies from 1593 to 3293 MWt, the drywell free volume varies from 131,000 to 164,000 ft³, the wetwell free volume varies from 98,000 to 131,000 ft³, and the suppression pool water volume varies from 62,000 to 135,000 ft³.

The design bases and the characteristics that are important to severe accident management are discussed below for the primary containment, the suppression pool, and the reactor building. Severe accident management, as defined in the NRC policy issue letter SECY-88-147 [38], 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 relevant to these two objectives are discussed.

3.1.1 Primary Containment

The primary containment for Peach Bottom has an internal design pressure of 56 psig, an external design pressure of 2 psig, and a design temperature of 281 °F. Its leakage rate is limited to less than 0.5% free volume per day at design pressure and temperature. The primary containment is maintained in an inerted state to reduce the possibility of hydrogen combustion. Containment inerting is achieved by maintaining a nitrogen rich containment atmosphere whose oxygen concentration is less than 4%. The primary containment is also designed to permit containment flooding (with water) to a level above the reactor core.

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 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 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

There is considerable uncertainty in estimating containment strength and failure mode. In the Severe Accident Risk Reduction/Risk Rebaselining Program (SARRP) a catastrophic drywell failure (a break area of 7 square feet) at a pressure of 117 psig is assumed in some of the radionuclide release calculations [15,16] and a wetwell failure at 159 psig is assumed in another more recent calculation [39]. The above pressure limits and the corresponding containment failure modes are based on analyses performed at Ames Laboratory [36] and at Chicago Bridge and Iron Co. [39], respectively.

Because of such uncertainties, probabilistic descriptions of containment failure pressure and failure mode are used in the NUREG-1150 study. Figure 3.2 shows the containment failure pressures for the five plants analyzed in NUREG-1150 [5]. The mean failure pressure shown for Peach Bottom's containment is 148 psig in low-temperature conditions. Since high drywell temperatures are expected to occur in most severe accident scenarios, Figure 3.2 also shows the effect of drywell temperature on containment failure pressure. The ranges of possible failure pressures (5th-95th percentile intervals) shown in Figure 3.2 for Peach Bottom are 120-174 psig at low temperature, 75-150 psig at 800 °F, and 6-67 psig at 1200 °F. These estimates were made for an "as-built" containment.

In the ranges of failure pressures shown in Figure 3.2, containment leakage (failure size of the order of 0.1 ft²) is most likely to occur near the lower end of the range (i.e., at pressures below mean failure pressure), and containment rupture (failure size in excess of 1.0 ft²) is most likely toward the upper end of the range. Furthermore, the probability of drywell failure increases with drywell temperature, and the drywell head is the most probable drywell leak location.

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. Although all Mark I plants have similar designs, their failure limits and corresponding failure modes depend on the details of the individual containment design and construction, and therefore will differ from plant to plant. Plant unique analyses are needed to determine failure limits and failure modes for individual plants.

3.1.1.2 Containment Fission Product Retention

In the absence of additional sources, the amount of fission products in the containment atmosphere will decrease with time by natural deposition processes, and consequently, the amount of fission products released to the environment will be reduced if containment failure is sufficiently delayed. Additional time also allows more radioactive decay to occur before FPs are released. Containment fission product sources are twofold: those arising from the degradation of the core materials in the reactor pressure vessel (RPV), and those resulting from the attack of the concrete floor by the molten core debris after vessel breach. With STCP modelling most of the release from the vessel occurs before or at vessel breach. After vessel failure and the start of core concrete interaction (CCI) the CCI will reach a peak and then diminish to a negligible level within a few hours [42]. 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 [37]. The models used in other severe accident codes may produce different CCI histories.

The fission product removal rate via natural deposition processes may be approximated by defining a removal rate constant such that the removal rate equals the product of the rate constant and the amount of fission products in the containment atmosphere. Assuming that natural deposition is primarily caused by gravitational settling of aerosols, the removal rate constant can be expressed as the product of a deposition velocity and the ratio of the deposition area and the containment volume. A rough estimate of the removal rate constant for Peach Bottom, using a deposition velocity derived in Reference 40, is 0.32/hr. This gives a fission product decontamination factor of 2 after 2 hours, and 25 after 10 hours.

3.1.2 Suppression Pool

The suppression pool 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 vent system (Figure 3.1). Main vents connect the drywell to a vent header in the suppression chamber airspace. Projected downward from the vent header are the downcomer pipes. These downcomers are 24 inches in diameter and terminate 4 ft below the surface of the pool. Typically there are 8 to 10 main vents and 48 to 120 downcomers (Peach Bottom has 96 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 vent system into the suppression pool, where steam is condensed and noncondensable gases are released to the suppression chamber airspace. Vacuum breakers are provided in the vent headers to equalize the pressure between the drywell and the suppression chamber if the pressure in the suppression chamber is greater than that in the drywell.

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 discharge pipes routed down the main vents into the suppression pool. The discharge end of the SRV piping is equipped with a sparger (T-quencher) to reduce the dynamic suppression pool loads resulting from SRV discharge and to promote uniform and stable steam condensation in the suppression pool. A typical Mark I T-quencher has an arm length of about 10 ft, an arm diameter of approximately 1.0 ft, and a total of about 1600 holes on each quencher, with a hole size of 1.0 cm in diameter.

The suppression pool is an alternate water source for the high pressure core injection systems (RCIC and HPCI), and the sole water source for the low pressure ECCS systems (LPCS and LPCI) and the containment spray (CS) systems¹. LPCI and CS are different operating modes of the RHR system and, as such, share components of the RHR system. The LPCI system is automatically initiated and controlled upon receipt of an injection signal. The other modes of the RHR system are manually initiated and controlled.

The energy deposited into the suppression pool during an accident can be removed from the suppression pool via the RHR heat exchangers. The ultimate heat sink for the RHR heat exchangers is provided by the high pressure service water (HPSW) system. For Peach Bottom, the normal HPSW water supply is from Conowingo Pond, a virtually unlimited source. When the normal service water source is not available an alternate water supply is furnished by the emergency heat sink (EHS) system. The suppression pool plays a very important role in fission product removal during a severe accident. It provides significant fission product scrubbing of any flows passing through it. Since it is the water source of many safety systems, pool conditions, such as water temperature and water level, affect the performance of the engineered safety features of these systems. A brief discussion of the role of the suppression pool in severe accident management is presented below.

¹. In Peach Bottom the high-pressure service water (HPSW) crosstie system provides an alternate water supply to the ECCS and the residual heat removal (RHR) system.

3.1.2.1 Suppression Pool Decontamination Factors

Suppression pool scrubbing is particularly effective for fission products produced in-vessel and released through the SRV spargers. The decontamination factor (DF) used in the NUREG-1150 analysis for in-vessel releases ranges from 1.2 to 4000 with a median value of 80 [5]. In comparison, the DF for flows passing through the downcomer vents is smaller, due to less submergence and larger bubble size. For downcomer flows DF values used in the NUREG-1150 analysis range from 1 to 90, with a median value of 10.

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 will discharge directly to the reactor building without suppression pool scrubbing if the containment fails 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 [20]. Reactor vessel depressurization is required if pool temperature exceeds the heat capacity temperature limit (HCTL) to (1) avoid exceeding the suppression chamber design temperature or the primary containment pressure limit (PCPL) [20], and (2) avoid excessive suppression pool boundary loads caused by unstable steam condensation from an SRV blowdown to a heated pool² [24, 41].

²A pool temperature limit was imposed by the NRC to preclude the occurrence of an unstable steam condensation load [41] and is part of the basis for the SP heat capacity temperature limit (HCTL) for Peach Bottom [24]. However, more recent data from small-scale tests showed that the dynamic pressure load for a quencher device, now used in Peach Bottom and other Mark I plants, may not cause any problem [58,59]. This loading condition is maintained in the following discussion because an official position on the applicability of the small-scale data for full scale plant conditions has not been published. Although small, the pressure does increase with decreasing pool subcooling, reaches a maximum at a pool subcooling of 25-30°K, and then decreases with further reduction of subcooling [58].

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 and can potentially lead to excessive loading from unstable steam condensation². A saturated suppression pool may cause the pumps that take suction from the suppression pool to fail from cavitation. 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 another source term issue addressed by expert elicitation in NUREG-1150.

3.1.2.3 Suppression Pool Water Level

The suppression pool water level is another EPG control variable [20]. 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 wetwell vent paths or the vacuum breakers between the drywell and wetwell.

Following the guidance of the BWR EPGs, the Peach Bottom Transient Response Implementation Plan (TRIP) provides specific directions and procedures to control suppression pool water level in an accident. It includes emergency procedures to add water to the suppression pool by the ECCS or the HPSW if the suppression pool water is depleted or insufficient, and procedures to remove water from the suppression pool if the water level is high. The systems mentioned in the Peach Bottom TRIP to remove water from the suppression pool include the torus filter pump (isolation bypass is required under accident conditions), which discharges suppression pool inventory to the main condenser hotwell or waste collector

tank, and the high pressure injection systems, which discharge suppression pool inventory to the condensate storage tank (CST) through the HPCI test return line [24]. 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. In Peach Bottom the torus dewatering facilities, which have a 1.2 million-gallon capacity storage tank and associated valving and piping, normally used for temporary storage of suppression pool water during inspections and/or modifications, can be used for the above purpose.

3.1.3 Reactor Building

The design and construction of the reactor building is similar in all Mark I plants (Figure 3.1). The lower levels of the reactor building, below the refueling floor, are reinforced concrete structures. Above this floor the building superstructure consists of metal siding and decking supported on a structural steel framework. The roof is usually constructed of steel sheeting. The internal design pressure of the reactor building is 0.25 psi and the design leakage rate of the secondary containment is 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. These blow-out panels are also designed to prevent a buildup in excess of a 3-psid during a tornado.

The reactor building 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 the off-gas stack. The height of the release point of the stack is 500 ft above ground level for Peach Bottom and varies from about 150 to 600 ft for other Mark I 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 Reactor Building Decontamination Factors

The reactor building provides additional fission product retention from natural processes such as aerosol deposition. The decontamination factor of the reactor building is primarily a function of the residence time of the transporting gases in the

building and thus depends on (1) the size and location of the primary containment break, (2) the ability of the reactor building to remain intact, (3) the magnitude and frequency of hydrogen burns, and (4) the driving force from the primary containment. The reactor building decontamination factors used in the NUREG-1150 analysis range from 1.1 to 10 with a median value of three for typical accident conditions.

There are significant uncertainties in estimating the effectiveness of reactor building 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

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 Peach Bottom is presented below.

The SGTS in Peach Bottom is common to both Units 2 and 3 and is located in a shielded room in the radwaste building between the reactor buildings. It contains parallel trains of particulate and iodine filters preceded by demisters, air heaters, and prefilters. It has a design flow rate of 10,500 cfm and is designed to maintain the reactor building at a negative 0.25 inch water gauge pressure when the reactor building is isolated [22]. (A maximum flow rate of 25,000 cfm can be reached with all three SGTS blowers running.) The high efficiency particulate air (HEPA) filters are water resistant and have a design temperature of about 200°F. Iodine is removed by activated charcoal beds, which have a typical design adsorption efficiency of 99 percent for elemental iodine and 95 percent for methyl iodide. The presence of adsorbed water on the charcoal surface will substantially affect this efficiency by reducing the surface area available for the trapping of volatile forms of radioactive iodine.

During a severe accident, the large amount of aerosols that can collect on the filter may plug the filter and reduce its flowrate. The pressure drop across the filter may even be sufficiently large to rupture it [15]. Although the HEPA filters of the SGTS may fail in a severe accident from aerosol plugging, some charcoal bed adsorption may still be maintained. Even in the case when both HEPA and charcoal filters fail, operation of the SGTS may be desirable because of the paths and release point associated with the SGTS. On the negative side, the operation of the SGTS blowers may reduce the residence time of fission products

in the reactor building and, in the event of a loss of system filters, accelerate the fission product release.

At Peach Bottom, the SGTS lines contain fire dampers which will close when the fusible links, which secure the damper in an open position, fail at a temperature of about 160°F. The release of the hot gases from the primary containment to the reactor building during a severe accident may trigger the closing of the SGTS fire dampers and prevent the use of the system for fission product removal later in the accident.

3.1.3.3 Reactor Building Fire Spray

The fire spray in the reactor building can scrub the fission products from the building atmosphere and thus mitigate fission product release. The fire spray system in the reactor building of Peach Bottom is a water curtain system designed to prevent fire spreading from one area to neighboring areas. Fire sprays start automatically at a high temperature (190 °F) or a high temperature rise rate (15 °F/minute). These temperatures may be reached when high temperature containment gas is discharged into the reactor building. Operation of the Peach Bottom fire spray system is terminated manually by shutting a gate valve located in a stairwell of the reactor building. Fire sprays may also be manually actuated at break-glass control stations in the reactor building [23].

The water curtain system at Peach Bottom is not as effective as the area spray system used in some other plants, e.g., Browns Ferry, for fission product scrubbing. The ability to operate the spray system manually is needed to make it an active fission product release mitigation mechanism. Although manual operation is available in Peach Bottom the locations of the control stations may prevent the system from being activated during a severe accident because of the hazardous environmental conditions at those locations. Actuating the fire spray system may increase the probability of hydrogen burns in the reactor building because of the removal of the steam from the building atmosphere.

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 [4] 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. Peach Bottom plant parameters are used for illustration.

3.2.1 Primary Containment Cooling and Water Supply

In Peach Bottom the primary containment cooling and ventilation system (a fan-cooler system) is used during normal plant operation to maintain the primary containment atmosphere at below 135 °F, and the residual heat removal (RHR) system is used during an accident for emergency cooling of the primary containment. The RHR system uses the suppression pool as its water source and can be operated in either the suppression pool cooling mode or the containment spray mode for containment cooling. A short description of these systems for Peach Bottom 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 water, is not available.

The Fan-Cooler System: The fan-cooler system in Peach Bottom consists of seven fan-coil units distributed inside the drywell. The two fans in each unit are motor-driven and the cooling water for the cooler is served by the chilled water system normally and by the reactor building cooling water system in the event of loss of offsite power. There are also four 15-hp fans and associated ductwork in the drywell to circulate the drywell atmosphere and to equalize the temperature throughout the drywell. These fans, as well as the unit coolers, trip automatically in the event of an accident, but can be manually restarted from the control room during an accident. The drywell coolers have a heat removing capacity of 2 MW.

The RHR System: The RHR system in Peach Bottom has two trains, with two motor-driven pumps for each train, to supply water to the primary containment. The RHR pumps take suction from the suppression pool and are powered by the emergency diesel generators if offsite power is not available. The heat is removed by the HPSW system through the use of any of the four RHR heat exchangers. Each heat exchanger has a heat removal capacity of approximately 20 MW. The containment cooling mode of the RHR system is manually actuated. Since it shares systems with the RHR core injection mode, its use is prohibited if core injection is demanded, and operator action is required to override the permissive core injection signal to initiate containment cooling.

Alternate Water Supplies for the RHR System: 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 (1) crossties with other plant

systems, (2) crossties with similar systems between different units in a multiple unit plant, and (3) outside the plant.

In Peach Bottom a crosstie with the High Pressure Service Water (HPSW) system is already available. A crosstie can also be made with the Fire Water (FW) system. Both HPSW and FW are usually attached to an extensive water supply. In the case of Peach Bottom, both systems take suction from an unlimited supply of pond water. In the unlikely event that pond water is not available, the HPSW system can take suction from the emergency heat sink system, which has a 3.7 million-gallon water storage reservoir, pumps, and a mechanical, induced-draft type cooling tower and is designed for events involving loss of normal plant heat sink. The fire water system in Peach Bottom has its own diesel engine driven pump as a backup to the electric motor driven pump, and, therefore, could supply water to the primary containment during station blackout when ac power from both offsite and standby diesel generators is not available.

The water from both the HPSW and the fire water systems is untreated and could plug the water supply system. There are other treated water sources that, although not normally available to the RHR system, can be used to supply water to it. These include the water from the Condensate Storage Tank (CST), the main condenser hotwell, the Demineralized Water Storage Tank, and the Refueling Water Storage Tank (RWST). Crossties may be arranged to make these water sources available to the RHR system [4]. The Peach Bottom emergency procedures include directions to the operators to lineup these systems for RPV water level restoration [24]. Since low pressure injection is provided by the RHR System, which also has a containment spray mode, water is thus available to the containment.

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. In cases where crossties are not available, they can usually be arranged without much difficulty, although preplanning is essential for their success. Peach Bottom has a crosstie of the HPSW system between Unit 2 and Unit 3. The two condensate storage tanks (CST) of the two units are frequently intertied to keep their water levels together. This mode of operation doubles the inventory of the CST when it is needed in a severe accident.

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. To provide makeup water to the emergency heat sink system for a postulated case with both reactors shutdown and the loss of normal heat sink, Peach Bottom has designed plans to transport water from offsite using truck trailers and to withdraw water from waterholes in the Susquehanna River or

Rock Run Creek using temporary hose lines and portable pumping equipment.

Table 3.3 presents information on some of the systems that can supply water to the primary containment in an accident, and Table 3.4 presents typical water capacities of some of the water sources discussed above [18,22].

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 Peach Bottom 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.

Electric Power: The Peach Bottom station has two independent sources of offsite power, one from an overhead 230 kV transmission line and the other from a 13 kv underground cable. The standby ac power is supplied by four onsite diesel generators (shared by the two units), which 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.

There are two independent 125/250-V dc power system for each unit. Each system is comprised of two 125-V battery banks, each with its own charger. 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 Peach Bottom, the dc power is expected to last at least 6 hours in a station blackout (SBO) event. Calculations show that this can be extended to about 12 hours if the operator is successful in reducing and isolating unnecessary loads [19].

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

Pneumatic Supplies: In Peach Bottom, pneumatic supplies are provided by the instrument air, the service air, and the instrument nitrogen systems via compressor operation. There are three air compressors for the instrument air and service air systems for each Peach Bottom unit, and a fourth air compressor common to both units. They deliver compressed air at 100 psig to the instrument air system for station instrumentation and controls and to the service air system for station services. Two (one instrument air and one service air) of the three parallel compressors normally supply all compressed air requirements for one reactor unit with the third compressor serving as a standby unit. The corresponding air systems of the two units can be crosstied [22]. The compressors require ac power to operate, and in Peach Bottom, the instrumentation air system is currently backed up by a portable diesel compressor [7].

Pneumatically operated devices located within the primary containment are normally operated by instrument nitrogen to avoid introducing additional oxygen into the inerted containment. The instrument nitrogen system is backed up by the instrument air system through a cross connection between these two systems. When low pressure is sensed in the instrument nitrogen receiving tank, the station instrument air system is automatically switched into service as a backup. Vital components, such as MSIVs and SRVs, are provided with accumulators to assure reliable function without compressor operation. A long-term, backup, safety grade, pneumatic supply is also provided to the ADS valve accumulators. The source of this pneumatic supply is a series of nitrogen cylinders located within the reactor building with a connection provided outside the reactor building for the installation of additional bottles, if required. For Unit 3 of Peach Bottom, a Safety Grade Instrument Gas (SGIG) System, which supplies pressurized nitrogen gas from the Containment Atmosphere Dilution System (CADS) tank and can be used as a back up to normal instrument air, is used to provide pneumatic supplies to the inflatable seals of the purge and vent valves of the containment atmosphere control system (CACS) and the valves of the suppression chamber-to-secondary containment vacuum breaker lines. For Unit 2, presently, the pneumatic supplies for the above valves are provided by the nitrogen cylinders located near the valves and will be provided later by the SGIG system after a tie-in to the hard-piped SGIG headers is implemented.

Pneumatic Supply Enhancement: NUREG/CR-5474 [4] has discussed strategies to enable emergency replenishment of the pneumatic supply for safety related air operated components. The options for additional air supplies include: service air system, diesel air compressors, and additional onsite storage of bottled gas systems. As discussed above, Peach Bottom, to some extent, has already implemented these options.

3.2.3 Primary Containment Spray System

The primary containment spray system is designed to keep the pressure and temperature loads on the primary containment to within their design basis limits. The spray headers in the drywell are located well up in the cylindrical part of the drywell, at the approximate elevation of the vessel head. Another set of spray headers are located in the wetwell. The containment spray system is part of the RHR system and takes its water supply from the RHR sources. In Peach Bottom a throttling valve located in the discharge end of one of the pumps in each of the two RHR trains can be used to throttle the RHR flow and therefore the CS.

In addition to its design function of containment pressure and temperature control, the drywell spray also is a significant severe accident management tool because of its ability to (1) remove fission products from the containment atmosphere, (2) provide water to the corium on the drywell floor to mitigate core-concrete interaction (CCI) and reduce the probability of drywell shell melt through.

Containment Spray for Fission Product Scrubbing: If given sufficient time, containment sprays are very effective in reducing airborne concentrations of fission product aerosols and vapors. The drywell spray removes airborne fission products by the mechanisms of impaction, interception, Brownian diffusion, diffusiophoresis, thermophoresis, and condensation. Although the efficiency of the spray in removing airborne fission products is sensitive to many parameters that have considerable uncertainties, a rough estimate of its effectiveness, through the definition of a spray removal rate constant and a collection efficiency, can be made.

The spray removal rate constant is defined to be proportional to the collection efficiency, the spray flow rate, and the droplet fall height, and inversely proportional to containment volume and droplet diameter [43], and can be expressed as $\exp(-at)$ where a is the removal rate constant. Using values typical for Peach Bottom and an average collection efficiency for particles of 2 microns in diameter, typical of STCP calculation, the spray removal rate constant is predicted to be 3.4/hr for a spray flow rate of 500 gpm with droplet diameter of 1 mm. This gives an effective decontamination factor of 10 after about 40 minutes of spray operation. Chemicals can be added to the spray water to enhance its fission product removal capability. The deliberate injection of sodium hydroxide at TMI enhanced iodine absorption by increasing the alkalinity of the water and is credited for the reduced iodine releases to the environment [44].

Effects of Drywell Spray on CCI: Operation of the drywell spray can flood the reactor cavity before core debris is discharged to the drywell floor and provide water continuously to the core

debris during subsequent core-concrete interaction (CCI). Potential beneficial effects of this water addition are (1) reducing fission product release by pool scrubbing, and (2) reducing and eventually terminating CCI if a coolable debris bed is formed in the cavity. Both issues involve substantial uncertainties. The fission product removal capability of a continuous overlying pool of water on core debris adds to the already significant fission product removal capability of spray scrubbing and becomes more important if the spray scrubbing efficiency is lowered due to insufficient flow rate. The important issue of core debris coolability determines not only the extent of ex-vessel fission product release from the core debris but also the severity of the challenges to containment integrity by noncondensable gas generation and structural erosion. Previous experiments performed at SNL indicate that the addition of water to the melt will not prevent CCI from continuing, while analyses performed by IDCOR indicate that the addition of water will result in quenching of the melt [9, 45]. Results from code analyses depend on model assumptions used in the computer code. Although both MAAP and MELCOR codes allow the modeling of a water layer on top of the core debris, the former assumes that the debris crust in contact with water will crack and allow water ingress, while the latter does not make such an assumption and the overlying water is modeled only as a heat sink. The water pool as modeled in CORCON therefore does not have as significant an influence as MAAP on the attack of concrete by the core debris. Additional data seems to be required to resolve this important issue.

The Effect of Drywell Spray on Drywell Shell Meltdown: The presence of water in the reactor cavity and the drywell floor may have a significant impact on drywell shell meltdown. Although it is generally agreed that the probability of drywell shell meltdown is lowered by the presence of water on the drywell floor, there is disagreement and uncertainty on its effectiveness [5]. A recent analysis (NUREG/CR-5423) indicates that the conditional probability of drywell shell failure is deemed highly unlikely if the drywell is flooded, while the probability of failure is significant if the drywell is dry. This report is currently under peer review and will provide a framework for future analysis, if needed, to resolve this important issue.

Possible Adverse Effects of Drywell Spray: Possible adverse effects of drywell spray include (1) unacceptable containment negative pressure load or unacceptable pressure differential between the drywell and the wetwell due to rapid drywell depressurization by spray operation, (2) containment deadening due to air in-leakage resulting from containment depressurization to below atmospheric pressure level and insufficient noncondensable gases in the containment, (3) loading conditions generated by the interaction of the core debris and the water in the reactor cavity at vessel breach, and (4) additional pressure load from steam generation caused by adding water to the hot core debris on a dry

concrete floor. 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.

Spray Flow Rates: The spray effectiveness as well as the extent of any adverse effects depend on the flow rate of the spray. Peach Bottom has procedures to initiate the drywell spray at 500 gpm to avoid rapid drywell depressurization [24]. Once spray has been initiated, the flow rate can be slowly increased to above 500 gpm while monitoring the drywell and wetwell pressures to assure they are within safe limit. However, under certain conditions, a very high flow rate is desirable. The containment spray system, as part of the high-capacity RHR system (4 pumps rated at 10,000 gpm each), has the capability of delivering very high flow rates. The effects of flow rate on droplet size and spray pattern in the containment may influence the spray's effectiveness for pressure reduction, fission product scrubbing, and corium cooling.

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 equals or exceeds a specified limit.

The venting system of Peach Bottom, typical of other Mark I plants, is described below. The important issues for a successful implementation of a venting strategy, the adverse effects of venting, and possible improvements for successful venting are also discussed.

3.2.4.1 Containment Venting for Peach Bottom

The vent paths identified in the Peach Bottom Transient Response Implementation Plan (TRIP) are reproduced in Table 3.5 and shown schematically in Figure 3.3. The vent paths are presented in Table 3.5 in a preferred order based on (1) the effect of suppression pool scrubbing, (2) vent path size, (3) the use of filtered vent paths, (4) the elevation of the release point, and (5) the use of a monitored release point [24]. An exception to the above general criteria is the use of the 2-in drywell vent path prior to the use of some of the torus vent paths because it is believed that the filtering of the exhaust by the SGTS compensates for the lack of suppression pool scrubbing.

The vent paths shown in Figure 3.3 include (1) the 18-in lines designed for the primary containment atmospheric control system (CACS), (2) the 2-in lines designed for the containment atmospheric dilution system (CADS), (3) the 6-in lines normally used as part of

the integrated leak rate test (ILRT) system, and (4) the 2-in lines normally used to transfer water from the drywell floor and equipment drain sumps to the radwaste system.

The 18-in lines are normally used for containment inerting/deinerting as well as for supplying makeup nitrogen, as required, to maintain an inert atmosphere during normal operation. Since these operations are normally performed under atmospheric pressure and temperature conditions, sheet metal ductwork is used for the part of the lines outside the two isolation valves in the reactor building. The exhaust lines are connected to the SGTS for cleanup prior to release to the atmosphere through the off-gas stack. All system valves are limited to a maximum opening position to assure that they would close if a LOCA were to occur during purging operations.

The 2-in lines are part of the CADS, which is a standby system used only in an accident to prevent the formation of a flammable mixture of oxygen and hydrogen in the containment. The system is designed for the expected oxygen and hydrogen generation rates following a LOCA. It is capable of supplying nitrogen to the containment at a pressure of about 30 psig and a flow rate of about 60 scfm. The 2-in exhaust lines of the system, which are selected for containment venting, discharge to the inlet plenum of the SGTS. The air-operated valves are controlled from the control room and the pressure control valves are preset to control downstream pressure to about 1 psig. All piping and valves in the CADS are designed to withstand the full 150 psig pressure of the stored nitrogen supply.

The ILRT lines are composed of steel piping and attached to the 18-in lines between the two isolation valves. The operation of the ILRT vent path requires access to the reactor building to close and open valves locally and to remove the flange outside the reactor building. The ILRT vent path discharges the containment atmosphere to outside the reactor building. The drywell sump drain path include two sump pumps which discharge into a common header containing two 2-in isolation valve in series.

The operation of the vent valves in Peach Bottom generally requires (1) ac power, (2) instrument air, and (3) bypassing an isolation signal. Wetwell venting also requires that the vent path is not submerged in the wetwell. The actions to minimize the effects of containment venting considered in Peach Bottom TRIP procedures include: (1) evacuating potential hazardous areas in and around the reactor building, (2) placing control room ventilation in the emergency mode to ensure that filtered outside air is supplied to the control room, (3) shutting down radwaste building ventilation to minimize the amount of fission products transported from the reactor building to the radwaste building, and (4) using only one train of the SGTS and putting the other train on standby.

3.2.4.2 Important Issues for Containment Venting

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 and possible hardware modifications to improve containment venting are discussed in the following.

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

As discussed in Section 3.1.1.1, there is significant uncertainty in the pressure capability of the containment. Although a Mark I containment is tested to 1.25 times the design pressure (typically 56 psig) and the use of this pressure assures containment structural integrity, a higher PCPL is desirable as well as practical because it is plausible that containment pressure capability is much higher than the design pressure. As a consequence, containment pressure limits of both 60 psig and 100 psig have been considered for Peach Bottom. Since containment strength may deteriorate as containment temperature increases, containment venting decisions may need to account for temperature also.

The other factors considered for PCPL determination arise from concerns of lack of driving pressure for air operated valves. Since the air pressure for the instrument air system is 100 psig (Section 3.2.2) a containment pressure higher than this value will make these valves inoperable. A higher air supply pressure for these valves will remove the above factors as limits in the determination of PCPL.

Areas and Flow Capacities of Vent Paths: The venting area required to maintain the containment pressure below the PCPL depends on the mass and energy input rates to the containment

atmosphere. Figure 3.4 shows the effective venting area required to keep containment pressure at a constant value for various energy input rates. (The curves shown in Figure 3.4 are based on isentropic flow of dry steam taken as an ideal gas and normal Peach Bottom containment airspace volume.) For a constant energy input rate, 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. Conversely, a larger vent area, while preventing a containment pressure increase, will also result in a faster and greater fission product release if the vent path cannot be reclosed in time.

The capacity of a vent path can be represented by the effective venting area shown in Figure 3.4. This effective area can be expressed as the product of the nominal opening area and factors accounting for the actual valve opening area, the friction loss in the flow path, and the effect of actual composition and real gas properties of the containment atmosphere. The actual valve opening area depends on valve design and valve arrangement and may be improved if the mechanisms that limit valve performance can be identified and improved.

The energy input to the containment atmosphere is a function of the type of accident sequence that is occurring and the progression of the accident. The flow capacities of 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.

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

Structural Capability of Vent Paths: Since the vent paths to be used for containment venting may not be originally designed for severe accident conditions, their structural integrity may be

questionable under venting conditions. In Peach Bottom, the ductwork of the vent path outside the primary containment will fail if the 18-in valve is opened at PCPL, and this will result in the release of the high pressure, high temperature, high radioactivity, and high hydrogen contents of the containment atmosphere directly to the reactor building [25]. On the other hand, the 2-in lines have been designed to take a higher pressure load and should remain intact if used for containment venting. However, the 2-in lines discharge to the inlet plenum of the SGTS, therefore, the effect of using these lines on the structural integrity of the SGTS needs to be investigated.

The most important valves in the vent paths are isolation valves and these are not expected to experience any structural damage at venting pressures. However, permanent structural deformation may develop during valve opening and closing operations because the operating conditions exceed valve design. This kind of structural damage will affect subsequent valve opening and closing operations.

Adverse Effects of Containment Venting: 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 (1) loss of plant safety equipment due to containment depressurization and suppression pool flashing, (2) reactor building contamination and resultant loss of safety related equipment or loss of accessibility for operator actions, and (3) fission product releases to the environment.

When the suppression pool temperature is high the loss of containment pressure from venting may cause the suppression pool to flash. This may result in failure of pumps taking suction from the pool due to lack of sufficient net positive suction head (NPSH). This mode of pump failure can be avoided by switching the pump suction from the suppression pool to an alternate water source before venting is started.

If the vent paths that have ductwork in the reactor building are used for containment venting, the failure of the ductwork will result in the discharge of the containment atmosphere directly to the reactor building. The discharge of this high temperature and highly radioactive atmosphere to the reactor building may damage safety equipment in the building or inhibit personnel access for possible repair and recovery operations. There is also the potential for a hydrogen burn in the reactor building. A hydrogen burn may damage the reactor building structure and reduce its fission product retention capability, or generate a puff release of fission products to the environment. Since these vent paths are usually the vent paths of high flow capacity they are generally required only for ATWS sequences when the energy input rate to the

containment atmosphere is high. It is therefore desirable to limit their use in other accident sequences if smaller vent paths can provide sufficient pressure relief.

Vent Path Operability Under Severe Accident Conditions :

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 reactor building, (3) the time and manpower available to perform the required venting operations. Most of the above requirements are dictated by the accident sequence that is occurring. Thorough investigation of vent path operability under various severe accident conditions, to identify problems and methods to surmount these problems, and clearly defined guidelines or procedures are essential for the success of containment venting. A few examples of the problems associated with performing containment venting and the general guidelines to alleviate these problems and improve containment venting practice are discussed below.

In Peach Bottom the vent paths are presented in the TRIP procedures in a preferred order. The vent paths are opened consecutively, following the preferred order, until enough vent paths are opened and containment pressure is controlled. This is desirable in a symptom based procedure. However, under certain severe accident conditions, it may be necessary, or beneficial, to open the vent paths in a different order based on additional plant instrumentation indications and the knowledge of the accident sequence in progress. For example, an ATWS event requires a large venting area in a short time duration, it may be desirable to open large vent paths first. On the other hand, the selection of a vent path for a station blackout event must consider the effect of the hazardous environmental conditions on plant personnel performing the manual operation of vent valves. The vent paths whose valves are shielded from high radiation areas should be identified and used under this condition. These event based vent selection criteria can result in a more effective accident management response for the intended accident sequence.

The opening of a vent path requires the defeating of isolation signals and, sometimes, additional work to line up the path lines. To assure a successful implementation of the venting strategy, operators could perform these preparatory actions before the actual venting pressure is reached. The time for starting the actual venting as well as this preparatory work may depend on the pressure and the pressure rise rate in the primary containment and the availability of personnel. The preparatory work is particularly important for station blackout events because manual opening of valves under hazardous environmental conditions is required.

Since venting may result in the release of radioactive materials, venting procedures should be accompanied by appropriate

evacuation procedures to minimize personnel exposure. This concern has been addressed in the Peach Bottom procedures, as discussed above.

Modifications to Improve Containment Venting: As part of the results of the Mark I containment performance improvement (CPI) program the NRC staff has recommended to the Commissioners the implementation of an improved hardened vent capability for the Mark I plants [46]. The modifications considered include (1) a hard pipe vent path, (2) dc operated isolation valves, and (3) a rupture disk in the vent path.

According to Reference 46, the hard pipe path would be installed from the outlet of an existing outboard wetwell vent containment isolation valve to the base of the plant stack. The pipe would be routed through a dc operated isolation valve and bypass existing ductwork and the SGTs. The vent path would also have adequate size to accommodate the required containment depressurization rate and be capable to withstand the anticipated environmental conditions of a severe accident. Furthermore, all isolation devices need to be capable of being operated without reliance on ac power, and a rupture disk would be installed in the vent path to prevent inadvertent containment venting and release of radioactivity.

Vent systems similar to the above description have been proposed by some Mark I plant licensees [47]. Plants whose vent systems are inadequate compared with the above requirement are expected to upgrade their systems in the near future. The containment venting system used in the assessment of containment and release management later in this report is assumed to have the above capabilities.

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 [32] and NUREG-0654 regarding radiological emergency response plans and preparedness [48]. 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 [48]. 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 reactor 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.

Table 3.6 presents an outline of the transfer of emergency response functions from the control room to the TSC and EOF under various emergency classes [49]. 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 to provide technical support to reactor operations, thus taking the primary responsibility for the containment and release management (CRM) of interest to this report. Nevertheless, the EOF assumes overall responsibility for accident management upon its activation.

As noted above, the TSC is activated during the Alert emergency class. The Alert emergency class is defined in Reference 48 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 [50]. The technical basis of an individual plant's EOPs are the BWR Emergency Procedure Guidelines (EPGs), Revision 4, prepared by the General Electric Company [20].

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. Figures 3.5 to 3.7 show the entry conditions and the control variables of these three guidelines. The EPGs are symptomatic guidelines: Operators' actions are based on the values of the control variables, e.g. suppression pool temperature (SP/T) in Figure 3.5, 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. The basis for the derivation of the limit curves and their adequacy for severe accident management are also discussed.

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 (Figure 3.5) for Peach Bottom are (1) suppression pool temperature above 95 °F, (2) drywell temperature above 135°F, (3) drywell pressure above 2 psig, or (4) suppression pool water level outside the range of 14.6' to 14.9' [24]. Peach Bottom uses a containment atmospheric dilution system (CADS) for containment hydrogen and oxygen control and does not have a hydrogen concentration entry condition in the primary containment control guideline in its TRIP. 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.

As shown in Figure 3.5, 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 both the drywell and the wetwell. According to the guidelines, the operator should first try to control the variables within predetermined limits using normal plant equipment. If this fails and containment conditions further degrade, the operator should then carry out the RPV control guideline to shutdown the reactor, to perform emergency RPV depressurization, and/or to take additional actions to secure containment integrity and equipment protection, actions such as containment venting and spraying or switching the suction source for emergency cooling system pumps.

Containment venting and containment spraying are two of the important actions the plant operation personnel can take to control primary containment conditions. In the BWR EPGs, containment venting is used both for containment hydrogen and oxygen control and for containment pressure control. Containment sprays are used in both drywell temperature and drywell pressure control. Both systems have also been identified as important plant features for severe accident management as discussed in Sections 3.2.3 and 3.2.4.

The procedures for using containment venting to control containment hydrogen and oxygen concentrations have been assessed in the Containment Performance Improvement (CPI) program [46] and were found to be acceptable. The use of containment venting to prevent containment overpressure failure requires the operator to vent the containment when containment pressure reaches the PCPL,

which is consistent with the requirement for severe accident management as discussed in Section 3.2.4.

The operation of containment sprays in the EPGs is more complicated. Containment sprays (both torus and drywell sprays) are primarily used to control containment pressure and temperature. However, drywell sprays can also be used to mitigate chugging loads by either depressurizing the drywell to terminate the vent flow or by moving more noncondensable gases to the drywell. Containment conditions for spray initiation and conditions under which drywell spray is inhibited are specified in the EPGs. The primary reasons for restricting drywell spray are to prevent an unacceptable pressure differential between the drywell and the wetwell, to prevent a negative drywell pressure from exceeding its design limit, and to avoid deinerting the containment atmosphere. In the BWR EPGs the spray initiation and restriction conditions are derived from engineering analyses utilizing best-estimate models. Since the EPGs are basically prepared for plant conditions varying from slightly abnormal to design basis LOCA, the assumptions used in these derivations may not be appropriate for a severe accident. For example, one particular important assumption that is used in the derivation of many initiation and restriction limits is the amount of noncondensable gas in the primary containment. In a severe accident the amount of noncondensable gases in the containment may be significantly higher than that used in the BWR EPGs due to noncondensable gases produced from both in-vessel zircaloy oxidation and ex-vessel CCI. The restriction in the EPGs may result in an unnecessary limitation on the use of drywell spray, an important severe accident management feature as discussed in Section 3.2.3.

The amount of noncondensable gases produced in the RPV during core degradation raises another concern. 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 noncondensable gases discharged to the suppression pool. The containment loads from these events after significant core degradation has occurred will be different than those used for design assessment. Consequently, containment damage may happen prior to the time expected in the EPGs if the loads under severe accident conditions are more serious. Since in a severe accident the SRV loading condition occurs only if the RPV is not depressurized, and the LOCA loading condition occurs only for a high pressure vessel breach, both loading conditions could be avoided by keeping the RPV depressurized.

3.3.2.2 Secondary Containment Control Guideline

The purposes of the secondary containment control guideline are to maintain the integrity of the secondary containment, to

protect the equipment in the secondary containment, and to limit radioactive releases to the secondary containment and the environment. As shown in Figure 3.6 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 a control variable exceeds a predefined maximum operating limit the operator is instructed to take actions to maintain the value within the limit and, if this fails, to isolate the systems that are discharging into the problem area. Finally, if the conditions further deteriorate, the operator should take action by entering the RPV control guideline, to shutdown the plant, or to carry out emergency RPV depressurization.

The approach taken in the EOPs [24] toward secondary containment problems is two-fold (1) actions to determine and isolate the cause of the problem and (2) at the same time, actions to minimize the effect of the problem. However, in a severe accident the plant condition may have deteriorated to a state such that the actions described in the EOPs are not practical. Available plant actions may have already been taken and failed to prevent core degradation, vessel breach, and containment failure, or an unisolable containment break may have developed. Under these severe accident conditions, the purposes of the secondary containment control remain the same as described above, but the approach and detailed procedures for operator actions may be different. Efforts to minimize fission product release utilizing the reactor building features discussed in Section 3.1.3 need to be emphasized.

3.3.2.3 Radioactivity Release Control Guideline

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

3.3.2.4 Additional Guidelines for Containment and Release Management

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

To focus the attention of the operating personnel on severe accident management a separate guideline specifically prepared for severe accident management, instead of modifying and extending existing EPGs to cover the whole range of severe accident conditions, may be desirable. Some of the later parts of the existing EPGs may be incorporated into the severe accident management (SAM) guideline for a smoother transition. A logical transition point from existing EPGs to SAM guidelines is when significant core damage has occurred. SAM includes both in-vessel and ex-vessel management, the present study of containment and release management (CRM) considers only the ex-vessel part of SAM. The relationship between the existing EPGs and the CRM guidelines is further discussed in Section 4, and Figure 4.3 shows the time phases for the application of different guidelines.

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) using available equipment and resources for accident management, and (2) directing actions to recover lost, or identify alternate, equipment and resources. As discussed in Section 3.3.1 the TSC is most likely activated and in control of plant emergency functions when CRM activities are demanded. The TSC has the capability to assess severe accident conditions and is suitable to manage the accident following more flexible guidelines. However, specific TSC personnel should be designated to take definite responsibilities to assure successful severe accident management.

3.3.3 Instrumentation, SPDS, and Environmental Qualification

The instrumentation required to assess the plant and its environs during and following an accident is described in Regulatory Guide 1.97 (Rev. 3) [27]. There are five types (A, B, C, D, and E) 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 variables that are most important to Mark I containment and release control, i.e., those of interest to the present study, are listed in Table 3.7, along with their required ranges and qualification categories.

Control Room instrumentation information is also required, in terms of the design of the Safety Parameter Display System (SPDS), as part of a nuclear plant Emergency Response Capability by the NRC in NUREG-0737 [32]. 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 safety status 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 type A, B, C, D, and E 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 may be lost. Contingency plans for obtaining plant information from local instrument taps rather than remotely in the control room may be of benefit in this case.

The three qualification categories are defined in Position 1.4 of Regulatory Guide 1.97 as follows: "In general, Category 1 provides for full qualification, redundancy, and continuous real-time display and requires onsite (standby) power. Category 2 provides for qualification that is less stringent in that it does not (of itself) include seismic qualification, redundancy, or continuous display and requires only a high-reliability power source (not necessarily standby power). Category 3 is the least stringent. It provides for high-quality commercial-grade equipment that requires only offsite power." For both Category 1 and 2 variables, the instrumentation should be qualified in accordance

with Regulatory Guide 1.89 [28]. 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 [29] are acceptable to Regulatory Guide 1.89 [28]. 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 temperature profile, an increase of 10% gauge pressure for pressure profile, and an increase of 10% in the time period the equipment is required to be operational. Figures 3.8 and 3.9 are typical temperature and pressure profiles for the qualification of instruments within the primary containment of both PWR and BWR plants [29,30]. IEEE Std 323-1974 also gives a typical integrated radiation dose of 26 Megarads and a spray exposure of demineralized water at a rate of 0.15 gal/min/ft². The instruments outside the primary containment are qualified for the expected environmental conditions, which may be less severe than those within the primary containment and are plant specific.

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

The availability of an instrument during a station blackout sequence depends on its power supply and seems to be plant specific. For example, in Peach Bottom the only containment variable that is available in the control room during a station blackout is suppression pool temperature [19], while in Browns Ferry both drywell pressure and the suppression pool water level instrumentation are powered directly or indirectly by batteries and are operational during a station blackout [10]. All plant 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 Peach Bottom (and most likely for other Mark I plants too), 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 for the additional plant status information not readily available in the control room. An independent power supply for plant parameters that are important to CRM such as that required by CPI for RPV depressurization may also be desirable.

Table 3.1 Domestic BWR Facilities with the Mark I Containment System

Plants Licensed for Power Operation	Licensee	Net MWe/un.
Browns Ferry Units 1,2, and 3	Tennessee Valley Authority	1065
Brunswick Units 1 and 2	Carolina Power and Light	790
Cooper Station	Nebraska Public Power District	760
Dresden Units 2 and 3	Commonwealth Edison Company	770
Duane Arnold	Iowa Electric Light and Power	540
Fermi Unit 2	Detroit Edison Company	1075
FitzPatrick	Power Authority State of New York	757
Hatch Units 1 and 2	Georgia Power Company	760
Hope Creek Units 1 and 2	Public Service Electric and Gas	1030
Millstone Unit 1	Northeast Utilities	650
Monticello	Northern States Power Company	540
Nine Mile Point Unit 1	Niagara Mohawk Power Corporation	610
Oyster Creek	GPU Nuclear Corporation	620
Peach Bottom Units 2 and 3	Philadelphia Electric Company	1050
Pilgrim	Boston Edison Company	670
Quad Cities Units 1 and 2	Commonwealth Edison Company	770
Vermont Yankee	Vermont Yankee Nuclear Power Corporation	500

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

	<u>Peach Bottom Unit 2 or 3</u>	<u>Ranges of all Mark I Plants</u>
Rated Thermal Power (Mwt)	3,293	1,593 - 3,293
Drywell Free Volume (ft ³)	159,000	131,000 - 164,000
Wetwell Free Volume (ft ³)	127,700	98,000 - 131,000
Suppression Pool Water Volume (ft ³)	122,900	62,000 - 135,000
Height of Elevated Release Point (Stack) (ft above ground)	500	180 - 600

Table 3.3 Systems for Primary Containment Water Addition (Peach Bottom)

System	Capacity	Head	Water Source	Power Source
RHR Containment Spray	4x10,000 gpm	542 ft.	Suppression Pool	AC (Diesel or Offsite)
HPSW	4x4,500 gpm	700 ft.	Conowingo Pond or Emergency Cooling Tower Basin	AC (Diesel or Offsite)
Fire Protection System	2,500 gpm (Motor-Driven Pump)	290 ft	Conowingo Pond	AC (Diesel or Offsite)
	2,500 gpm (Backup Diesel-Driven Pump)	290 ft		Own Diesel

Table 3.4 Water Source Capacities (Peach Bottom)

System	Capacity
CST	156,000 gal.
Hotwell	90,000 gal. (minimum)
RWST	450,000 gal.
Clarified Water Storage Tank	200,000 gal.
Demineralized Water Storage Tank	50,000 gal.
Torus Dewatering Tank	1,200,000 gal.
Emergency Heat Sink Water Storage Reservoir	3,700,000 gal.
Portable Pumps	Variable
Municipal Water Systems	Variable

Table 3.5 Containment Vent Paths (Peach Bottom)

2" Torus Vent Line Through SGTS
2" Drywell Vent Line Through SGTS
6" ILRT Line From the Torus
18" Torus Vent to SGTS
18" Torus Purge Supply Line
6" ILRT Line From the Drywell
18" Drywell Vent to SGTS
18" Drywell Purge Supply Line
2-3" Drywell Sump Drain Line

Table 3.6
Transfer of Emergency Response Functions From the Control Room (CR) to the Technical Support Center
(TSC) and the Emergency Operations Facility (EOF)
(Reference 49)

Emergency Response Functions	Notification of Unusual Event	<u>Emergency Class</u>		
		Alert	Site Area Emergency	General Emergency
Supervision of reactor operations and manipulation of controls	CR	CR	CR	CR
Management of plant operations	CR(TSC)	TSC	TSC	TSC
Technical support to reactor operations	CR(TSC)	TSC	TSC	TSC
Management of corporate emergency response resources	CR(TSC,EOF)	TSC(EOF)	EOF	EOF
Radiological effluent and environs monitoring, assessment, and dose projections	CR(TSC,EOF)	TSC(EOF)	EOF	EOF
Inform Federal, State, and local emergency response organizations and make recommendations for public protective actions	CR(TSC,EOF)	TSC(EOF)	EOF	EOF
Event monitoring by NRC regional emergency response team*	(CR)	TSC(EOF)	TSC & EOF	TSC & EOF
Management of recovery operations	CR(TSC,EOF)	TSC(EOF)	EOF	EOF
Technical support of recovery operations	CR(TSC,EOF)	TSC	TSC	TSC

Note: (CR), (TSC), (EOF), or (TSC, EOF) indicates that activation of this facility (or the performance of this function) is optional for the indicated emergency class.

**Table 3.7 Some Accident Monitoring Variables Important
to Containment and Release Control**

Variable	Range	Category
Drywell Pressure	0 - design pressure	1
Primary Containment Pressure	-5 psig to design pressure	1
Primary Containment Pressure (Post Accident)	-5 psig to 3 times design pressure for concrete; 4 times design pressure for steel (For detection of potential for or actual breach)	1
Drywell Atmosphere Temperature	40° F to 440° F	2
Suppression Pool Water Level	Bottom of ECCS suction line to 5 ft. above normal water level (For detection of reactor coolant pressure boundary breach)	1
Suppression Pool Water Temperature	40° F to 230° F	2
Containment & Drywell Hydrogen Concentration	0 to 30 Vol. %	1
Containment & Drywell Oxygen Concentration	0 to 10 Vol. %	1
Drywell Drain Sumps Level	Top to bottom	1
Primary Containment Area Radiation (High Range)	1 R/hr. to 10 ⁷ R/hr.	1
Reactor Building Area Radiation	10 ⁻¹ R/hr. to 10 ⁴ R/hr.	2
Variables Monitoring the Operation of Containment and Safety Systems, e.g., ECCS, RHR, Containment Sprays	Water level, temperature, and flow rate	2 or 3
Variables Monitoring Area Radiation and Airborne Radioactive Materials Release	Radiation level, radioactivity per unit volume, and vent flow rate	2 or 3
Variable for Meteorology Conditions	Wind direction, speed, vertical air, temperature	3
Onsite Accident Sampling Capability	Hydrogen and oxygen contents	3

MARK I CONTAINMENT (Peach Bottom)

Figure 3.1 Schematic of the Containment Design for the Peach Bottom Plant.

Figure 3.2 Containment Failure Pressure. (Reference 5)

Figure 3.3 Peach Bottom Purge and Vent Valve Arrangement. (Reference 25)

Figure 3.4
Venting Area and Venting Pressure
For Various Reactor Power Levels

PRIMARY CONTAINMENT CONTROL

Figure 3.5
(Reference 20)

SECONDARY CONTAINMENT CONTROL

Figure 3.6
(Reference 20)

RADIOACTIVITY RELEASE CONTROL

Figure 3.7
(Reference 20)

Figure 3.8
Typical Temp. Qualification Profile

Figure 3.9
Typical Pressure Qualification Profile

4. STRATEGY IDENTIFICATION

A significant amount of information on severe accidents is available from the research efforts guided by NRC's Severe Accident Research Plan (SARP) [51] and the analyses found in the industry sponsored Industry Degraded Core Rulemaking Program (IDCOR) [18]. The data base provided by the above research efforts is reviewed to identify the challenges a Mark I containment could face during the course of a severe accident, the mechanisms behind these challenges, and 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 described first. This is followed by a discussion of the challenges, the mechanisms behind the challenges, and the mitigating strategies during different phases of a severe accident. The safety objective tree summarizes the results of this identification effort (Figure 4.15). Besides being used for challenge and strategy identification, the CRET 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 are also discussed.

4.1 Containment and Release Event Trees

The containment and release event trees used in the present investigation consist of three trees covering the different phases of a severe accident. The background for the development of the CRETs, their general structure, and the application of the CRETs for challenge and strategy identification, strategy effectiveness quantification, and as a CRM tool for decision making are discussed below.

4.1.1 Containment Fission Product Release Profile and Accident Progression Event Trees (APET)

The amount and timing of the release of radioactive materials to the environment (source term) determines the offsite consequence of a severe accident. The source term, in turn, depends on the containment failure conditions developed during the accident, which in NUREG-1150 [5] are grouped into accident progression bins (APBs). Figure 4.1 shows the conditional probability of the APBs at Peach Bottom for the plant damage states (PDSs) that contribute significantly to the total plant core damage frequency (CDF).

Figure 4.1 presents a general picture of the containment fission product release profile for Peach Bottom. It identifies the high consequence APBs and their probability of occurring for each PDS. The objective of severe accident management is to reduce the probability of the high consequence APBs through operator intervention. This requires the understanding of the events that lead from a PDS to the APBs.

Accident progression event trees (APET), or containment event trees (CET), have been used in NUREG-1150 to track the progression of the potential accident from the time core damage is imminent through core concrete interaction (CCI). The APET organizes and quantifies the important characteristics of the core damage process, the effects of operator actions, the challenges to the containment building, and the response of the building to those challenges, and thus provides the information required for the identification of CRM strategies. Figure 4.2 shows the relationship between the PDS, the APET, and the APBs. As indicated in Figure 4.2 the APET serves as a transfer function between the PDS and the APBs, and changes to any element of the APET, by either system modification or operator actions, will change the APBs, i.e., the release profile.

The APET considers in a chronological order events that are important to accident progression and the associated containment building response. There are 145 top events (or questions) in the Peach Bottom APET, addressing issues regarding the entry states of the accident, the phenomenological events, the reactor vessel breach conditions, the containment system's survivability, and the containment failure modes. The APET can be broken into the following 5 time periods: (1) the initial period, including questions 1 through 22, determines the conditions at the beginning of the accident, (2) the core vulnerable period, including questions 23 through 46, addresses the progression of the accident during the period the operators are attempting to avert core damage, (3) the core damage period, including questions 47 through 69, determines the progression of the accident from the beginning of core damage to just before vessel breach (4) the vessel breach period, including questions 70 through 109, addresses primarily the loads accompanying vessel breach and the containment structural response to these loads, and (5) the late period, including questions 110 through 145, determines the progression of the accident during CCI [8].

The APET presents a detailed description of the containment events that may happen during a severe accident. It contains information from many sources, e.g., detailed computer accident simulations and panels of experts providing interpretations of available data. The APET represents a source of up-to-date knowledge of containment events and provides information needed for the development of accident management strategies.

4.1.2 Simplified Containment Event Tree for CRM

The APET as described above can be used as a basis to identify both the challenges to containment integrity as well as important operator actions to mitigate the consequences of these challenges. To focus attention on containment and release management (CRM), containment and release event trees (CRETs) which consider only those elements that are important to CRM are constructed for this report. The CRETs consist of three event trees covering different accident progression phases. Figure 4.3 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).

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. 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, existing EPGs or in-vessel activities may be carried out concurrently with release management activities.

Elements important for severe accident management include (1) accident initiation for the early CRET, i.e., PDSS, or plant status for the late and release CRET, (2) efforts for recovery or the locating of alternate equipment and resources, (3) challenges to containment integrity, (4) operator actions to mitigate these challenges (strategies), and (5) containment and release responses. Tables 4.1 through 4.3 list the key elements of the early, late, and release CRET, respectively, and Figure 4.4 presents a simplified CRET for the early phase of station blackout (SBO) sequences.

Figure 4.4 uses the top events listed in Table 4.1, but is simplified in that individual challenges and strategies are not presented. It is shown here to demonstrate the application of the CRET in SAM. Figure 4.4 indicates that, in general, containment integrity is not challenged in the very early phase of an SBO sequence (before core melt). However, containment integrity could be challenged in this time period if there is a recovery of core injection without a concurrent recovery of containment heat removal

(CHR). The accident will then progress like a TW (loss of CHR) sequence and the containment can fail by overpressure if it is not vented or CHR is not recovered in time.

Most likely, containment integrity will still not be seriously challenged in the early phase of an SBO sequence, between core melt and vessel breach. However, the combination of SP boundary loads and containment pressure loads, or the use of small capacity but insufficient core injection could result in a challenge during this phase. Containment integrity could also be challenged at vessel breach. Since the challenges associated with vessel breach usually result in rapid containment failure, strategies to mitigate these challenges would need to be implemented before vessel breach to be effective.

A detailed CRET, with all the challenges and strategies included in the tree structure, and with appropriate probabilities assigned to the individual elements of the tree, can be used to quantify the effectiveness of strategies [17].

The same tree structure can also be used during an accident to predict the probable results of accident progression for accident management. The actual existing plant status information and best estimate probability data (e.g., the probability of recovery) could be used in quantifying the trees to provide a prediction of the effect of various CRM strategies on accident progression. Probable consequences resulting from the implementation of the various CRM strategies could then be estimated by a simple consequence code using the containment release profile predicted by the CRET. Meteorology and emergency offsite activities would be known at this time, and could be factored into the analysis. Such CRET based predictions can provide important and reliable accident management information for CRM decision making.

Detailed examinations of various accident sequences using the CRET tree structures for challenge and strategy identification are discussed in the following sections. The examinations reveal the challenges and their mechanisms during different accident phases. Strategies that need to be carried out before these challenges take effect can then be identified.

4.2 The Identification of Challenges, Mechanisms, and Strategies

The CRETs are used to examine some important accident sequences in order to identify the challenges, the mechanisms behind these challenges, and the strategies which can mitigate these challenges. The discussion below is divided into the time periods of the three CRETs.

4.2.1 The Early CRET

The PDS defines the initial and boundary conditions of a severe accident. It identifies the initiating event, the availability of safety features and resources, and other conditions such as containment isolation status and suppression pool bypass. As shown in Figure 4.1 the PDSs that contribute significantly to the total core damage frequency (CDF) of Peach Bottom, if only internal events are considered, are SBO, LOCAs, ATWS, and Transients. A more detailed break down of the important PDSs, their mean CDFs (MCDF), and their percentage contribution to the total mean CDF (TMCDF), for Peach Bottom from NUREG-1150 analyses, is shown in Table 4.4. It shows that SBO and ATWS are the most important PDSs and contribute 47% and 43%, respectively, to the total mean CDF of the plant.

Different PDSs will cause different challenges to the containment at different times of the accident. Table 4.5 shows the timing of key events for accident sequences associated with the above PDSs and other sequences such as the TW sequence. It shows that containment failure could occur at different times in different sequences. The values shown in the figure are from calculations by the Source Term Code Package (STCP) [15, 16, 52] and are typical for accident progression without any operator intervention. The timing of major accident events depends on many factors. For example, the availability of core injection from the control rod drive hydraulic system (CRDHS), the existence of stuck-open relief valves (SORVs), operator actions such as containment venting, and computer code model assumptions such as those that affect zirconium oxidation could all change the timing of the key events.

The containment pressure load presents the most significant challenge to containment integrity in this phase of an accident. Figure 4.5 presents a typical containment pressure time history during an accident. The containment is pressurized in the very early phase, before core melt, by the steam generated from the primary system. The rate of containment pressure increase is slow if core power is at decay heat level and fast if the reactor fails to scram as in the ATWS sequences. Containment pressurization after core melt is caused by the steam and hydrogen generated in the RPV via decay heat and zirconium oxidation in the early phase.

4.2.1.1 The Very Early Phase of the Early CRET

The challenges to containment integrity during the very early phase, before significant core melt has developed, include suppression pool boundary loads and containment pressure loads. The challenges, and the mechanisms and strategies associated with these challenges are presented in Table 4.6. The mechanisms that may cause significant suppression pool (SP) boundary loads include (1) SRV air clearing with higher than normal SP water level, (2)

SRV steam condensation with pool temperature greater than the SP temperature limit (Section 3.1.2.2), and (3) downcomer vent chugging with low noncondensable gas content in the drywell. The mechanisms that cause significant containment pressure loads include (1) loss of pressure suppression capability either due to high SP temperature or SP bypass and (2) inadequate containment heat removal (CHR). The drywell temperature may exceed its design value in some accident sequences but it will not reach a value that challenges containment integrity in this very early phase. 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 are expected to be applicable during this phase of an accident (Section 3.3.2). 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. The mechanisms and the strategies presented in Table 4.6 are discussed in more detail below.

SP Boundary Loads: Suppression pool (SP) water level will exceed its normal value if water sources other than the SP are used for core injection or containment spray (e.g., HPSW system by crosstie, Section 3.2.1), and this may happen in any accident sequence. A high SP water level will result in a longer water slug in the SRV discharge line to be cleared before the air in the SRV line is discharged to the SP during SRV actuation. The air volume discharged into the SP will be more compressed and could cause significantly larger dynamic loads on SP boundary and structures. The clearing of the water slug may also damage the SRV line, the SRV discharge quencher device, or its supporting structures. Since these loads increase with increasing RPV pressure, they can be reduced or eliminated by RPV depressurization if SP water level control is unsuccessful.

The discharge of the primary system steam to a SP at high temperature may result in a significant SP boundary load from unstable steam condensation (Section 3.1.2.2). The pool temperature limit currently imposed by the NRC for condensation oscillation loads is about 200 °F for high SRV discharge rates [41]. The temperature of the SP may exceed this value in sequences where SP cooling is lost, e.g., TW, or insufficient, e.g., ATWS. This condition may also happen in SBO sequences if core injection is maintained for a long time but CHR is not recovered in time. (Core injection may be maintained during SBO by the use of an

alternate water supply, by manual operation of the high pressure injection system, or by extending the dc power supply.) Since the condensation stability limit is dependent on the SRV discharge rate, timely RPV depressurization can be used to reduce the SRV flow rate and thus prevent this challenge.

Another SP boundary load that may cause containment structural damage is the chugging load generated from steam condensation in the SP when the drywell atmosphere is discharged to the wetwell through downcomer vents in LOCA sequences. The noncondensable gases originally in the drywell will be carried to the wetwell with the drywell to wetwell flow as primary system inventory is discharged to the drywell. Higher chugging loads can occur when noncondensable gas concentration in the drywell is very low. The drywell spray can be used to condense the steam in the drywell to move some of the noncondensable gases from the wetwell back to the drywell through vacuum breakers. The spray may reduce pressure sufficiently to terminate the vent flow from the drywell to the wetwell and thus eliminate the chugging load. Any subsequent chugging arising from the repressurization of the drywell will be mitigated by the presence of the additional noncondensibles.

Containment Pressure Loads: As shown in Table 4.5, detrimental containment pressure loads may occur from either a loss of SP vapor suppression or inadequate CHR. SP vapor suppression is lost in cases of SP bypass, low SP water level, or high SP temperature. Containment sprays can be used to condense the steam in the containment atmosphere during SP bypass, and RPV depressurization can be used to remove the source of heat input to the SP under high SP temperature or low SP water level condition.

In ATWS sequences, where containment energy input exceeds the designed CHR capability, and in TW sequences, where the designed CHR capability is not available the containment failure pressure can be reached. SBO sequences where core injection is maintained via in-vessel recovery actions, as in the cases discussed above, will result in a situation similar to the TW sequences if CHR is not recovered in time. The pressure-time history in the very early phase depends on the sequence. The rate of pressure increase will be large in ATWS sequences, and as a consequence, containment failure pressure could be reached in a short time (e.g., within one hour as shown in Table 4.5). At the other end of the spectrum, the pressure increase will be slow in a TW sequence and containment failure pressure will only be reached after a long time even if CHR is not restored. As shown in Table 4.5, the time for containment failure in a TW sequence could be more than 30 hours after accident initiation.

Containment venting can be used to relieve containment pressure and prevent containment failure. Since containment venting is used here before core damage begins, the release of fission products to the environment will not be significant. The

use of a hardened vent path, as recommended in the CPI program [46], eliminates the possibility of losing essential safety equipment located in the reactor building due to the discharge of high temperature steam from the primary containment. Other venting issues are discussed in Section 3.2.4.2.

4.2.1.2 The Early Phase of the Early CRET

The early phase of a severe accident covers the time period between the onset of core melt to slightly after vessel breach. 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.

The challenges to containment integrity during this time period include the SP boundary load and containment pressure and temperature loads before vessel breach, and the more severe loading conditions, which have the potential to fail the containment very rapidly, at or immediately after vessel breach. These severe loads include those associated with high pressure melt ejection (HPME) during vessel blowdown, the loads caused by steam explosion as the molten core debris interacts with the water in the reactor cavity, and thermal attack of the drywell shell by the molten core debris shortly after vessel breach. The challenges, mechanisms, and strategies for the early phase are outlined in Table 4.7.

Containment venting is suggested as a strategy in many cases in Table 4.7. Venting through the wetwell is desired because the containment atmosphere could be highly contaminated in this phase of a severe accident. As shown in the Table 4.7, this phase is divided into two time periods, the time period before vessel breach and the one after vessel breach. A detailed discussion of the challenges, mechanisms, and strategies in the two time periods are presented below.

4.2.1.2.1 Before Vessel Breach

SP Boundary Load: If the RPV remains at high pressure when core melt is in progress the reactor pressure will reach the SRV set point and SRV actuation will begin. The mass of noncondensable gases discharged into the SP following SRV actuation is much greater than that originally in the SRV discharge line, used as the basis for the design assessment load. As a result, 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: Containment pressure loads during this time period result from mass and energy addition to the containment atmosphere and hydrogen combustion. Mass and energy addition arise from the heat and gases generated in the RPV from decay heat and fuel cladding oxidation (principally zirconium oxidation in this time phase). Containment failure during core melt is predicted to happen in the AE sequence (large LOCA with loss of core injection) in Table 4.5. It should be noted that the predicted pressure in the containment during core melt is sensitive to the core slumping scenario assumed and the fuel cladding oxidation fraction calculated from the model. Considerable uncertainty exists. The predicted containment pressure rise during core degradation for the AE sequence in Table 4.5 is about 106 psi with 36% zirconium oxidation [15]. This value is used in NUREG-4700 as the pessimistic estimate, and 45 psi and 16 psi (5% zirconium oxidation, typical of MAAP code calculations) are used as the central and the optimistic estimate, respectively. A containment pressure failure in this time period probably is not very likely. However, the operation of small scale coolant injection systems, e.g., CRDHS, if not sufficient to terminate core degradation, will prolong this time period, result in additional zirconium oxidation and hydrogen generation, and increase the probability of containment failure during this period.

Containment cooling and containment sprays can be used to remove energy and steam from the containment atmosphere and thus reduce the temperature and pressure rise in the containment. However, the above systems cannot remove noncondensable gases from the containment atmosphere. Containment venting is required to maintain containment pressure if the use of containment cooling and containment sprays is not sufficient to keep the pressure below the acceptable limit.

Containment Temperature Load: A very high drywell temperature is predicted for the AE sequence of Table 4.5 [15]. 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.

Hydrogen Combustion: Hydrogen concentration in the containment atmosphere is influenced by the amount of zirconium oxidized during core degradation. With 10% zirconium oxidation the amount of hydrogen generated in the RPV will be about 300 lb-moles for Peach Bottom. This is about half the amount of noncondensable gases initially in the containment (660 lb-moles). A significant portion of this hydrogen will be released to the containment (about 50% for high RPV pressure cases and 100% for low RPV pressure

cases) and the containment hydrogen concentration would exceed its combustible limit. Since the Mark I 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 vacuum breaker system between the suppression chamber and the reactor building. This can occur due to low containment pressure, as a result of containment venting or containment sprays (Sections 3.2.3 and 3.3.2.1).

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 Mark I containments that have a containment atmospheric dilution system (CADS) the CADS can be used to supply nitrogen to the containment for containment venting and purging. However, the CADS is a medium pressure (30 psig), low capacity (60 scfm) system (Section 3.2.4.1). Its effectiveness in containment hydrogen control under severe core degradation conditions is limited. For plants such as Peach Bottom which has the CADS but does not have a hydrogen control entry condition in their EOPs (Section 3.3.2.1), it may be desirable to incorporate the hydrogen control procedures of the EPGs into the EOPs.

The systems that can be used to supply gases to the containment are the primary containment atmospheric control system (CACS, Section 3.2.4.1) and the CADS discussed above. The CACS has a higher capacity (5 containment volume changes per 9 hours at Peach Bottom) than the CADS but operates at a lower pressure (designed for atmospheric pressure). Containment venting is required to reduce the containment pressure before these systems can be used. In addition to the above function, containment venting reduces the initial containment pressure before hydrogen combustion, and, as a consequence, reduces the peak containment pressure following hydrogen combustion.

The containment spray can 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 have the potential of reducing the probability of detonation [53]. However, there is considerable uncertainty in droplet size and fog density resulting from fog formation in the containment during severe accidents. Moreover, large-water-droplet sprays tend to increase flame speeds by promoting mixing in lean hydrogen-air mixtures and cause peak pressures to be closer to the adiabatic, constant-volume values. Furthermore, containment sprays will change the containment atmosphere composition by removing steam (a diluent) from the containment atmosphere and may thus

increase the probability of hydrogen 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.

4.2.1.2.2 After Vessel Breach

As shown in Table 4.7, the loads that challenge containment integrity after vessel breach include (1) SP hydrodynamic loads, (2) containment pressure loads, (3) transient pressure differential loads on the RPV pedestal created during HPME, and (4) drywell shell meltthrough. 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 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 an 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 noncondensable 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, and (3) ex-vessel steam explosions. Early wetwell venting to reduce the initial containment pressure before vessel breach is suggested in Table 4.7 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 3.2.4.2.

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 is predicted for the fast ATWS sequence in Table 4.5. This is the result of the combined effect of high containment pressure before vessel breach (about 80 psia) and the containment pressure rise from the high pressure blowdown (greater than 52 psi). For the slow SBO sequence in Table 4.5 the predicted total pressure is 129 psia. This is slightly less than the projected containment failure pressure of 132 psia [16]. However, there is considerable uncertainty regarding the pressure capability of the containment, as well as the pressure rise at vessel breach and containment failure cannot be totally ruled out.

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' sensible heat can be transferred rapidly to the atmosphere. The metal in the dispersed debris can react chemically with the oxygen or steam in the containment atmosphere (an exothermic reaction) and release more energy and noncondensable 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 (slump-type failure or flow-type 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 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 [5].

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 and drywell floor, which would help prevent drywell shell meltthrough 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.

Steam Explosion: When the molten core debris contacts water, rapid heat transfer between core debris and water may produce steam explosions. However, the shallow water pool in Peach Bottom reactor cavity (less than 18 inches) is unlikely to generate significant steam explosions. Nevertheless, the pressure spike due to rapid steam generation, although expected to be moderate in Peach Bottom (an estimated pressure rise of about 30 psi has been used in the draft NUREG-1150 calculation [9, 54]), will add to the pressure load that already exists in the containment from other sources and may result in the failure pressure being reached.

The pressure loads from core debris-water interaction can be avoided by removing the water in the reactor cavity sumps (approximately 2000 gallons at Peach Bottom) and preventing the use of any system that can add water to the reactor cavity, e.g., drywell spray. These actions are in direct conflict with other strategies that require the flooding of the reactor cavity or the use of the drywell spray. Coupled with the low probability for significant steam explosions in shallow water, this strategy (eliminate water) is not likely to be implemented. Current information indicates that the expected moderate pressure increase from core debris-water interaction can be accepted in light of the potentially more significant benefit of having water in the reactor cavity.

As in other cases that may cause rapid containment pressurization, early wetwell venting can be used to reduce initial containment pressure and thus the impact of pressure rise from steam explosion.

Transient Pressure Load on RPV Pedestal: The volume beneath the RPV (reactor cavity) is small and restricted (Figure 3.1). The communication between this volume and the drywell is one or two walkways and the hydraulic control unit passageway. 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 (greater than 10 ft²) 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. Since this load is caused by a high pressure RPV blowdown it can be prevented by RPV depressurization.

Drywell Shell Meltthrough: Drywell shell meltthrough (DSM) is the principal cause of early containment failure predicted in NUREG-1150 for Peach Bottom. Among the significant PDSs shown in Figure 4.1, the mean probability of DSM is over 30% for LOCAs and Transients and over 40% for SBO and ATWS. Two containment failure modes are possible when hot core debris contacts the steel drywell shell: the containment may fail by creep rupture due to the combined effects of high containment pressure and high local structure temperature, or the drywell shell may be penetrated by the debris. The probability of DSM is influenced by many factors of great uncertainty, and has been the subject of considerable controversy among severe accident experts. Essentially half the experts whose opinions were elicited in the NUREG-1150 analysis believe that containment failure would occur and half that it would not occur [5]. However, it is generally agreed that the probability of DSM is lowered by the presence of water on the drywell floor, and a more recent analysis shows that the conditional probability of DSM is small if the drywell is flooded (Section 3.2.3). Therefore flooding the reactor cavity via the drywell spray or some other means of water addition should reduce the probability of DSM significantly.

4.2.2 The Late CRET

As defined in Figure 4.3, a severe accident enters the late phase after vessel breach but before containment failure. The sudden change in RPV and containment conditions associated with vessel breach indicates the beginning of the late phase. Failure of the RPV may result in a sudden increase in containment pressure

and a sudden decrease in RPV pressure. The radioactivity in the drywell atmosphere may also show a sudden increase, since prior to vessel breach the discharge of the fission products to the containment atmosphere was through the SRV lines and the suppression pool, while after vessel breach the discharge goes directly to the containment atmosphere. This increase in radioactivity will depend on the specific scenario and is likely to be more pronounced for transients than for LOCA's. Indication of vessel breach may also be inferred from in-vessel instrumentation.

Containment Conditions of the Late Phase: Containment pressure and temperature at the beginning of this phase depend on the previous accident progression. For the accident sequences presented in Table 4.5, containment failure is predicted to happen in the late phase only for SBO sequences. Figure 4.5 shows a typical pressure time history for such accident sequences. The pressure spike occurring at the late stage of the early phase is due to a high pressure vessel blowdown, and the pressure drop occurring immediately afterward is due to condensation and convection heat transfer to containment structures. There will be no pressure spike if the vessel is breached at low pressure, and containment pressure at the beginning of the late phase would be lower. Typical initial containment pressure predicted for the late phase of SBO sequences is about 100 psia for high pressure vessel breach and 40 psi for low pressure vessel breach. Containment pressure continues to rise after vessel breach, and may fail the containment if mitigating actions are not taken in time. The pressure rise in the late phase is primarily caused by the energy and gases released from decay heat and CCI.

Typical drywell and wetwell temperature time histories, as calculated by the source term code package (STCP), are shown in Figure 4.6. Before vessel breach, the primary system inventory is discharged to the containment through the SRVs and the suppression pool. After vessel breach, the temperature in the drywell rises rapidly as the hot primary system inventory is discharged directly to the drywell atmosphere. Drywell temperature immediately after vessel blowdown may be greater than its design value (281°F for Peach Bottom), but is not expected to be severe enough to threaten containment integrity. Further into the late phase, drywell temperature continues to rise due to energy and gases generated from decay heat and CCI. Drywell temperature could exceed 1000°F a few hours after vessel breach if containment cooling systems are not activated and CCI continues. Wetwell temperature stays much lower than drywell temperature during CCI, and is not expected to cause any structural problem. However, a high SP temperature may damage equipment taking suction from the SP (Section 3.1.2.2).

Core-concrete interaction (CCI): CCI is the most important mechanism for containment loading in the late phase. As the high temperature core debris falls into the reactor cavity after vessel breach, the molten core debris starts to heat and decompose the

structural concrete. The steam, carbon dioxide and other oxidants released from the decomposing concrete will react with the metallic constituents in the corium and generate a significant amount of noncondensable and combustible gases and release the chemical heat of reaction into the corium pool. 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 (Figures 4.5 and 4.6). 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 spread and geometric configuration of the core debris on the floor, and the composition and material properties of the structural concrete.

Figure 4.7 shows an example of concrete attack by CCI, typical for cases predicted by the CORCON model of the STCP. Concrete erosion in the radial direction essentially ceases after the metallic and the originally heavy oxide layers invert, at about 150 minutes after vessel breach, while concrete penetration in the vertical direction continues. Concrete erosion has an important influence on containment performance: The radial progression of concrete erosion may weaken the reactor pedestal and result in vessel motion including tearout of piping penetrations through the drywell wall and consequently drywell failure. The downward progression of the concrete erosion may result in basemat melt-through and subsequent fission product release. Both of the above failure modes are expected to take a considerable time to occur and analyses show that the likelihood of this mechanism of failure is small for the BWRs analyzed, in part because other mechanisms are likely to result in failure earlier in the accident. For Peach Bottom, the depth of the basemat of the containment, directly under the vessel, is so great that it is unlikely the basemat would be penetrated before the occurrence of other failure modes [5].

The addition of water to the molten core debris (corium) on the floor may affect the progression of CCI. In the CORCON model, the corium and water are treated as separate layers with film boiling at the interface. Water on top of the corium affects the corium surface heat transfer but has little effect on concrete erosion. Conversely, in the IDCOR approach, the corium is assumed to be quenched in most cases where water exists above the corium [55]. There exists some potential for the corium to be quenched as a coolable debris bed is formed in the reactor cavity and cooled by a continuous source of water. Additional discussion on the effect of an overlaying water pool on CCI have been presented in Section 3.2.3.

Noncondensable and/or combustible gases, as well as steam, are produced from CCI. These gases can cause a significant pressure load on the containment and present some potential for combustion. Figure 4.8 shows the gases generated from CCI, typical for those

predicted by STCP. The total amount of noncondensable gases generated by CCI five hours after vessel breach is about 3000 lb-moles, or about four and half times the noncondensable gases originally in the containment (660 lb-moles). Since the temperature of these gases can be very high (about 3000°F for the debris pool shown in Figure 4.9), release of these noncondensable gases to the containment atmosphere will cause severe pressure and temperature loads on the containment. Heat transfer between the corium and containment atmosphere further increases the containment pressure and temperature load. Figure 4.9 shows the temperature of the corium/concrete interface predicted by STCP (the predicted corium temperature is similar to the interface temperature), the temperature surge at about 100 minutes after vessel breach is the result of oxidation of the zirconium in the corium.

The composition of structural concrete has a significant effect on the amount of noncondensable gases generated during CCI. The gas generation presented in figure 4.8 is based on the limestone concrete in Peach Bottom, which has a much higher ablation temperature and gas evolution rate than does the siliceous concrete used in some other plants. Consequently, the amount of gases generated from CCI with siliceous concrete would be less than those shown in Figure 4.8.

Table 4.8 lists the challenges, mechanisms, and strategies in the late phase of an accident. The primary challenges in this phase are containment pressure and temperature loads and the erosion of containment structures.

Containment Pressure Loads: The sources of containment pressure loads are the heat and noncondensable gases generated during CCI and 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 4.1.1.2.1. Containment pressurization via heat and noncondensable gases and the associated strategies are discussed in the following.

Containment Cooling: As discussed above, the noncondensable gases generated from CCI can result in a significant containment pressure. The pressure increase is proportional to both the amount of noncondensable gases released to the containment atmosphere (mass addition) and the amount of energy transported to the containment atmosphere (energy addition). This includes the energy carried by the gases as well as the energy from atmosphere/corium interface heat transfer. Containment cooling can be used to remove energy from the containment atmosphere and thus reduce pressure. If the containment temperature can be maintained close to its normal value, then the pressure rise from the mass addition of five hours of CCI would result in a pressure 4.5 times the atmospheric pressure (Figure 4.8). On the other hand, the same pressure rise would occur in less than two hours if the temperature of the released noncondensable gases is 2000°F. Since the mass addition

rate slows after three hours (Figure 4.8), containment cooling may significantly extend the time to containment failure.

Containment Venting: Containment cooling can remove energy from the containment atmosphere but cannot prevent eventual containment failure if generation of noncondensable gases by CCI continues. Containment venting is capable of removing both mass and energy from the containment atmosphere and will be needed to prevent failure if CCI continues. Since the containment may be highly contaminated at this time wetwell venting should be used. However, since the required venting rate may be small during the late phase of an accident, the small venting area available with the operation of the SGTs may also be sufficient to achieve successful pressure control.

Corium and Reactor Cavity Flooding: The effect of water addition on CCI progression is still uncertain. However keeping the corium submerged has the potential of slowing down, and eventually terminating CCI. Adding water to the drywell to quench the corium is an important strategy. While adding water to the corium on a dry floor may cause an initial pressure increase due to steam generation, the increase is in general expected to be moderate.

Flooding the reactor cavity before vessel breach may help the core debris to form a coolable debris bed and result in a subsequent quench of the core debris. Water in the reactor cavity would also affect the phenomenological events occurring immediately after vessel failure, e.g., steam explosion and drywell shell melt-through, which have been discussed in Section 4.1.1.2.2.

Drywell Spray: Drywell spray can be used to keep the corium submerged and to reduce the rate of pressure increase in the containment by removing energy from the drywell atmosphere. Figures 4.10 and 4.11 show the effects of drywell spray on drywell pressure and temperature during CCI. The data presented in Figures 4.10 and 4.11 are from STCP calculations of an SBO sequence 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 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. The Mark I containment has been designed to withstand the flooding loads even if it is filled to a level above the reactor core (Section 3.1).

As water is added to the SP, the gases in the wetwell airspace will be displaced to the drywell through the vacuum breakers

between the drywell and the wetwell until the 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 (127,700 ft³ for Peach Bottom, 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. For the in-vessel strategy of flooding the containment to cover the bottom head of the RPV, calculations have been carried out for Browns Ferry (a plant similar to Peach Bottom in size) [56]. The water required to cover the bottom head is estimated to be about 200,000 ft³ in that analysis.

Containment flooding is part of the RPV control guideline in the BWR EPGs (Contingency #6) [20], and as such it may be initiated early in an accident (as compared with most of the containment strategies discussed here). To flood the containment, the EPGs call for the suppression pool makeup system to rapidly add a large quantity of water to the containment, and for all available systems that take suction from sources outside the containment to deliver water to the containment. Contingency #6 of the EPGs describes in detail the systems, water sources, and procedures to be used to fill the containment.

The 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 Peach Bottom, 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 HPSW system (two pumps, 4500 gpm for each pump, Table 3.3) is available for supplying water to the SP, the time required to flood the containment up to the RPV bottom head (assume 200,000 ft³, or 1,500,000 gallons) is approximately three hours. The time will be significantly shortened if all four RHR pumps are available for containment flooding (4x10,000 gpm, Table 3.3). However, the water available for containment flooding using the RHR pumps may be limited by the capacity of water sources for these pumps, e.g., the combined capacity of the CST and the hotwell is only about 250,000 gallons (Table 3.4), 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 200,000 ft³ free air space by water is more than two thirds of the original air space volume of the whole containment (The combined drywell and wetwell air space is about 290,000 ft³ for Peach Bottom, 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 half its original value, and the pressure will correspondingly be increased by a factor of two.

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.

Drywell Temperature Load: As shown in Figure 4.6, 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 (e.g., the containment fails at 1200°F for the SBO sequence analyzed by IDCOR [18]) or a combination of temperature and pressure loading (as discussed in 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. As shown in Figure 4.11, drywell spray is very effective in controlling the drywell temperature.

Basemat Melt-through or RPV Pedestal Erosion: Because of their small likelihood, basemat melt-through or RPV pedestal erosion does not present a significant risk. However, they may pose a long term threat if other mechanisms that could fail the containment sooner are avoided by corrective strategies. 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.

4.2.3 The Release CRET

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 reactor building, as indicated by the quite extensive arrays of area radiation, temperature, and flood monitoring systems in this building. In Peach Bottom, alarms will be set off in the control room when the monitored parameters exceed their setpoints. Their values depend on the measuring location and can be as low as 5 mr/hr for area radiation monitors (ARM), 105°F for area temperatures, and 6" above the floor for area flooding measurements [24].

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 fission product (FP) release after containment failure.

Table 4.3 shows the key CRM elements in the release phase, which include knowing the plant status, the potential for recovery, the challenges to FP release, possible mitigating actions, existing or expected offsite conditions, including both meteorology and emergency response conditions, and potential release consequences. Possible source terms from a severe accident are shown in Figure 4.12. Important factors influencing the source terms include the mode and location of release, the various decontamination processes, and the pathways for FP release.

To facilitate in the discussion of the challenges and mechanisms later in this section and the strategy assessment in Section 6, the characteristics of FP release under severe accident conditions are described first. This is followed by a detailed description of the challenges, mechanisms, and strategies for FP release control (Table 4.9).

4.2.3.1 The Characteristics of FP Release

There are three major barriers that must be breached before significant amounts of FPs are released to the environment: the fuel cladding, the RCS pressure boundary, and the primary containment. As the temperature of the core increases during an

accident, the fuel cladding is breached first and FPs are dispersed in the RCS. Depending on the scenario, and the integrity of the RCS these FPs will be discharged through the SP by SRV actuation or directly to the primary containment without the benefit of SP scrubbing. A part of the FPs that are airborne in the containment atmosphere would be released to the outside of the containment should the containment lose its integrity, either by containment failure (CF) or by containment venting.

Airborne FPs in the containment atmosphere can also originate from CCI, from high pressure melt ejection, from revolatilization of the FPs in the RCS, and due to late releases of iodine from the water pools. These mechanisms of FP release will be discussed in more detail in the sections discussing FP release challenges and mechanisms.

Fission Product Groups: Nine FP groups are used in the NUREG-1150 analyses to characterize the FP release profile, or source term. The quantity of a source term is typically characterized by the fraction of the core inventory of the FP groups that are released. Table 4.10 shows the nine FP groups and their inventories for Peach Bottom. The radionuclides in the same FP group in Table 4.10 have a similar chemical form and release characteristics.

The radionuclides in different FP groups have different effects on offsite consequences. Table 4.11 presents the early and late effect weights of the different source term groups used in the NUREG-1150 study for Peach Bottom. When the offsite consequence considered is an early effect (early illness, early fatality, or individual risk of fatality) the release is expressed in terms of the fraction of core inventory of the iodine group which would have been released to obtain the same 24 hour average bone marrow and lung dose effects. For late effects (latent cancer, population dose) an equivalent release for the cesium group is used for consequence calculation [54].

The effects of different FP groups can also be expressed in terms of the noble gases released to the environment (Table 4.12) [44]. Noble gases have been used as the benchmark in Table 4.12 because they would almost certainly be released for all important severe accident release conditions. Table 4.12 shows the release fraction for each group that could produce a whole body dose at 5 miles equivalent to that contributed by exposure to 100% of the noble gases for two release times, 2 hours and 24 hours after reactor shutdown. Table 4.12 indicates the effect of the short half-life of the noble gases on the whole body dose. The dose for 100% release of noble gases after a 24-hour delay is decreased by a factor of 5 below that for a 2-hour delay.

Airborne FPs in the Primary Containment: In general, the FPs released to the environment are carried by the flow of the

containment atmosphere to the outside of the containment (except for an ISL event). The amount of FPs in the containment atmosphere is therefore an important parameter affecting the release of radioactivity to the environment. Figure 4.13 [40] shows the general features and time variation of the FPs released to the primary containment from the RCS, early in the accident from core degradation (FRCS) or later from revolatilization (FREV), from HPME, from CCI (FCCI), and from the release of iodine from water pools late in the accident (FLI). Table 4.13 shows the fractions of FPs released to the containment by two of the most important sources, the release from the RCS (FRCS) and the release from CCI (FCCI), and their approximate release durations. The values presented in Table 4.13 are based on the highest STCP calculated fractions from the accident sequences in the assigned category. They are higher than the median values, but are within the ranges of distributions, used in the NUREG-1150 analyses.

4.2.3.2 Challenges, Mechanisms, and Strategies

The challenges, mechanisms, and strategies during the release phase are shown in Table 4.9. For some of the challenges, the conditions that could lead to FP release can be detected before actual release occurs and actions to mitigate these challenging conditions can be taken either before or after containment failure (CF). For other challenges, diagnostic and mitigating actions are possible only after FP release has started. A detailed discussion follows.

FP Concentrations in the Containment Atmosphere: The fission products in the containment atmosphere will be discharged to outside the containment, either to the reactor building or directly to the environment, after containment failure. One effective means to reduce FP release to the environment is to reduce the amount of airborne FPs in the containment.

Although the FPs in the containment atmosphere will not cause any FP release problem if containment integrity is maintained, the removal of FPs from the containment atmosphere is important when containment failure is imminent or has occurred. The reduction of the FPs in the containment atmosphere before CF is also important because in some cases it is more effective to reduce airborne FP while the release into the containment is taking place. The reduction is also proportional to the duration of the action, and early initiation (before CF) is again important.

As discussed in the previous section, fission products are introduced into the containment by various mechanisms and pathways during various time frames, and strategies to reduce FP concentrations in the containment atmosphere should therefore be designed accordingly.

Release From the Vessel: The release of the FPs from the vessel to the containment is one of the important sources considered in NUREG-1150 for potential FP release to the environment (Figure 4.12). Table 4.13 presents the approximate fractions of the various FPs released from the vessel to the containment (FRCS) for severe accident conditions. In general, almost all of the noble gases are released, and significant fractions of the more volatile radionuclides (I and Cs groups) will also be released. The release of Te is more uncertain but is expected to be significant, while the release of other less volatile FP groups will be small. The release duration, from the start of core damage to vessel breach, predicted by the STCP is typically 1.5 hours.

For Peach Bottom and other Mark I plants the release of the FPs from the vessel to the containment will pass through the SP (by SRV actuations) as long as RCS integrity is maintained, and as a result, the FPs discharged to the containment atmosphere may not be significant. As a result, the impact on environmental consequences may not be important if the SP decontamination factor is high (Section 3.1.2.1). The release from the vessel will be more important for those sequences which bypass the SP, e.g. large LOCA.

Since the SP is very effective in the removal of radionuclides in the form of aerosols or soluble vapors and some of the most important radionuclides, such as iodine, cesium, and tellurium, are largely released from the RCS during the in-vessel release period, it is important to ensure that any in-vessel FP release to the containment should pass through the SP (by SRV actuation). For sequences where fission products are released directly to the drywell, without passing through the SP, e.g., a LOCA, drywell spray can be used to remove fission products from the drywell atmosphere (section 3.2.3).

FP Release During Vessel Breach (VB): 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 may result in significant aerosol generation and fuel fragmentation. The fission products discharged to the containment atmosphere include those originally in the RPV gas 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.

FP Release From CCI: CCI begins when the high temperature core debris falls into the reactor cavity and starts to attack the structural concrete. The volatile radionuclides (I and Cs) that have not been released in-vessel would be released during CCI. The release of other FP groups depends on the progress of CCI, which involves great uncertainties (Section 4.2.2). Among the factors that influence the magnitude and timing of the ex-vessel release

are the composition of the concrete beneath the reactor vessel and the composition and temperature of the debris as it is released from the vessel. Table 4.13 presents the fractions of the fission products released during CCI (FCCI), obtained from STCP calculations. Although the attack of the structural concrete could last a long time, as already discussed in Section 4.2.2, the duration of the significant FP release from CCI would only last a limited time according to STCP. In typical STCP calculations, which assume a single, instantaneous release of corium, it would only take a few hours for the corium to cool down sufficiently so that further FP release is negligible. Typical time durations for significant CCI release are approximately two to three hours after vessel breach.

The CCI release fractions calculated by STCP and shown in Table 4.13 are generally higher than the median values used in NUREG-1150. The uncertainty distributions obtained from the experts for NUREG-1150 are broad. The important factors considered in NUREG-1150 for the Peach Bottom CCI release are the amount of zirconium in the debris and the presence of water in the reactor cavity. The exothermic reaction of zirconium oxidation will release a significant amount of heat, which will increase the temperature of the corium and enhance the release of fission products. On the other hand, flooding of the reactor cavity can eliminate the CCI release of radionuclides if a coolable debris bed is formed, or can significantly attenuate the release from CCI via scrubbing in the overlying pool of water [5].

As discussed above, flooding the reactor cavity before VB and continuously adding water to the core debris after the core debris falls to the drywell floor can moderate the progression of, and may eventually terminate, CCI (Section 4.2.2). The existence of a water pool on top of the corium will reduce FP release to the containment atmosphere by pool scrubbing even if CCI is not terminated. The drywell spray can be used to both add water to the corium and scrub airborne fission products from the containment atmosphere (Section 3.2.3). Containment flooding as discussed in Section 4.2.2 will keep the corium submerged and could be used as a long term strategy.

FP Revolatilization from the RCS: As a result of the decay heat, fission products deposited on the surfaces within the RCS can be re-evolved after vessel failure. This process can be risk-significant because the revolatilized fission products are released later in a severe accident and may not undergo SP scrubbing (e.g., released after both RPV and containment have failed). 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.

Among the radionuclide groups, only iodine and cesium are likely to have significant revolatilization. Tellurium may also re-evolve, but the amount involved would be much smaller. The median values used in NUREG-1150 for BWR plants are about 10% for iodine, 5% for cesium, and 0% for tellurium. Although the median value (the value with 50% cumulative probability) is zero for tellurium, the value with 95% cumulative probability is about 20%,

indicating that the potential for substantial revolatilization cannot be ruled out.

Fission product revolatilization will be reduced if the temperatures in the RCS and the drywell are low. Adding water to the vessel and initiating containment cooling may achieve the above purpose. Adding water to the RPV will also scrub fission products from the RCS and thus reduce their release. Drywell spray can be used to both reduce containment temperature and scrub airborne fission products. 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.

Late Release of Iodine From Water Pools: Late release of iodine from the suppression pool or the water pool on the drywell floor represents a long term challenge to release control. One problem associated with late release of iodine is the formation and release of volatile forms of iodine, e.g., elemental iodine, which would not be removed by pool scrubbing or containment structure deposition. 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.

In the NUREG-1150 study, four cases of late release of iodine are considered: a subcooled SP, a saturated SP, a flooded drywell, and a limited supply of water in the pedestal that mostly boils away. The median values of late iodine release are 0.1%, 0.5%, 50% and 80% for the above four cases, respectively.

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.

Driving Force for FP Release: 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: After CF, fission products in the containment atmosphere will be released to outside the containment, either to the reactor building or directly to the environment. 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, [34]). The total pressure-dependent leak area for a Mark I containment, including that from the drywell head, the drywell equipment access hatches, the personnel airlock, and the CRD removal hatch, is estimated in NUREG-1037 [34] to be less than 0.004 in² at 62 psig and approximately 12 in² at 82 psig. Among these potential leak areas leakage at the drywell head, caused by the stretching of the drywell head bolts is one of the most important early CF modes identified in NUREG-1150. Such a failure will result in a direct leak path from the containment to the refueling bay area of the reactor building (RB), bypassing the suppression pool and most of the reactor building.

Since the volumetric flow rate from a 1 in² leak area is approximately one containment volume per day for Peach Bottom, 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, fission product amounts released to the environment are reduced. Additional release reduction may be achieved by selecting a wetwell vent path that provides a greater reactor building 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. Reactor building 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 reactor building or directly to the environment. The probability of this failure mode is very low for Mark I containments because a failure 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 an interfacing systems LOCA (ISL) event (V sequence), the radioactive material in the RCS can escape directly to the reactor building or the environment. This occurs when a failure of the pressure isolation valves (PIVs) between the high pressure and low pressure systems results in the rupture of the low pressure piping from excessive pressure. For BWRs, the probability of containment bypass from an ISL is assessed to be quite small by NUREG-1150 [5]. The probability of an ISL for Mark I containments can be further reduced significantly by the provision of a leak testing program for the PIVs, which is in place at Peach Bottom [57].

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 trapping some of the 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 RB to the Environment: According to the BWR EPGs, the secondary containment HVAC should be isolated and the SGTS initiated if the secondary containment HVAC exhaust radiation level exceeds the isolation setpoint (20 mr/hr). The system will also be isolated upon receipt of a plant isolation signal (Section 3.1.3). The SGTS then provides a filtered release of the RB atmosphere to the environment at an elevated location. Direct

release to the environment is prevented because the SGTS maintains the RB at a negative 0.25 inch water gauge pressure.

The combination of RB isolation and SGTS operation can significantly reduce the release of non-noble gas radioactive materials to the environment if the flow from the primary containment (or the primary system if there is a bypass) to the reactor building 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 RB will increase, and leakage directly to the environment, as well as failure of the RB blow-out panels, or even structural damage to the RB or the SGTS may result. Even under conditions where substantial leakage from the RB develop, 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.

Figure 4.14 shows the volumetric leak rate (in terms of the Peach Bottom containment volume indicated in Table 3.2) from the primary containment as a function of containment failure size (in terms of an equivalent diameter for the leak area). The curves presented in Figure 4.14 are based on an isentropic flow model for choked flow of air at various containment temperatures, and present reasonable approximations to those predicted by the STCP. According to Figure 4.14, a leak area of 6 inches will result in a volumetric leak rate of about 2 containment volumes per hour. Since the SGTS has a maximum capacity of 25000 cfm, or 5.2 containment volumes per hour, it can handle the flow of a containment leak from a six inch leak area if the containment is leaking at a pressure not significantly greater than atmospheric pressure¹. At a higher pressure, the containment volumetric flow rate that can be managed by the SGTS is proportionally smaller. For a containment leaking (or venting) at a pressure of 100 psig (about 8 times atmospheric pressure), a volumetric leak rate of about 0.6 containment volumes per hour, would approach the capacity of the SGTS. This corresponds to the leak rate of a 3-inch leak area, and larger leak areas would result in a pressure rise in the RB and the potential consequences mentioned above.

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

¹The volumetric flow rates presented in Figure 4.14 are upper bound values. The flow rates will be less if the flow is not choked (containment pressure less than about twice the atmospheric pressure). The flow rate of an unchoked flow also depends on the pressure ratio.

amount of aerosols collected on the filter increases. The HEPA filter could be ruptured when the pressure drop across the filter reaches a 12-inch water pressure. This could result from a collection of about 50 lbm of aerosols [15].

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 Mark I containment performance improvement program, a hardened vent path bypassing the SGTS (Section 3.2.4.2) is recommended 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 RB 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 because its fire dampers close, as discussed in Section 3.1.3.2.

If a significant amount of fission products is released to the reactor building, the RB fire spray system can be used to scrub fission products from the RB atmosphere and consequently reduce the release of fission products to the environment. A discussion of the fire spray system and its use during a severe accident is contained in Section 3.1.3.3.

4.2.3.3 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 similar to those presented in Tables 4.1 to 4.3, 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.

4.3 The Safety Objective Tree

The results of the above strategy identification effort are summarized in the safety objective tree shown in Figure 4.15. 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 failure, it becomes important to minimize the amount of fission products released under these circumstances. Figure 4.15 was constructed according to the process defined in Section 2.2 and the results of strategy identification presented in Tables 4.6 through 4.9. It systematically defines the challenges to the overall safety objectives for a Mark I containment, identifies safety functions that need to be preserved to meet the objective and lists the specific challenges found in a Mark I 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 4.15, 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 4.2 Late CRET Top Events

1	Plant Status After Vessel Breach	LPS
	Instrument, Equipment, and Resource Availability	
	Containment Status (Pressure and Temperature)	
	Core Concrete Interaction (CCI) Status	
2	Late Recovery	LRV
3	Late Challenges	LCH
	Containment Pressure Loads	
	Drywell Temperature Loads	
	Basemat Melthrough or Pedestal Erosion	
4	Late Operator Actions	LOA
	Containment Cooling	
	Wetwell Venting or Small Area Venting with SGTS	
	Corium and Reactor Cavity Flooding	
	Drywell Spray	
	Containment Flooding with Drywell Venting	
5	Late containment Failure	LCF

Table 4.3 Release CRET Top Events

1	Plant Status After CF	RPS
	Instrument, Equipment, and Resource Availability	
	Containment Failure Mode (Size & Location)	
	Containment Failure Time (Early or Late)	
	RPV (Early Failure) or CCI (Late Failure) Status	
	Containment Conditions (Pres., Temp., Fission Products)	
	Suppression Pool Thermal Condition	
2	Recovery	RRV
3	Challenges	RCH
	Fission Products in Containment Atmosphere	
	Driving force for Fission Product (FP) Release	
	FP Release from Containment to Outside containment	
	FP Release from RCS to Outside Containment	
	FP Release from Reactor Building to Environment	
4	Release Operator Actions	ROA
	Containment Actions	
	RPV Depressurization	
	Identify and Isolate Leak Area	
	Containment Spray	
	Wetwell Venting	
	Leak Area Flooding	
	Containment Cooling	
	Suppression Pool Cooling	
	Corium and Cavity Flooding	
	Containment Flooding	
	Reactor Building Actions	
	Fire Spray	
	SGTS	
5	Offsite Conditions	ROC
	Meteorology conditions	
	Emergency Response Conditions	
6	Release Consequences	RCS

Table 4.4

Plant Damage States for Peach Bottom (Reference 8)

<u>PDS Number</u>	<u>PDS Name</u>	<u>MCDF (1/yr)</u>	<u>PDS % TMCDF</u>
1	LOCA, RHR	2.6E-07	5.8
2	Fast transient SORV, RHR	2.2E-07	4.9
3	Fast transient SORV, No RHR	6.1E-09	0.1
4	Fast Blackout	2.1E-07	4.7
5	Slow Blackout	1.9E-06	42.0
6	Fast ATWS, SLC	3.0E-07	6.7
7	Slow Blackout	1.1E-07	2.4
8	Fast ATWS, SLC	1.5E-06	33.0
9	ATWS, SORV	4.4E-08	1.0
	ATWS		
	ATWS, LOSP		

Table 4.5 Typical Timing for Different Accident Phases of Various Accident Sequences
(STCP Calculations)

Accident Sequence	Very Early Phase		Early Phase		Late Phase
	Accident Initiation (hr)	Onset of Core Melt (hr)	Vessel Breach (hr)		
Slow SBO (Note 2)	0	10	12		CF (15) (Note 1)
Fast SBO (Note 3)	0	1	3		CF (5)
Slow ATWS (Note 4)	0	CF (1)	4		
Fast ATWS (Note 5)	0	0.5-1	2		
AE (LOCA) (Note 6)	0	0.2 CF (0.5)	CF (2)		
TW (Note 7)	0	CF (30)	45	50	

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. Both ac and dc power are lost at accident initiation.
4. In slow ATWS containment failure causes the loss of core injection systems.
5. Core injection is lost before containment failure. Containment fails at vessel breach. Core melt may begin in less than 0.5 hour if core injection is lost at accident initiation.
6. This is a large LOCA with the loss of core injection. The time for core melt may be from 1 to 2 hours in an intermediate size LOCA. The time of CF depends on Zircaloy oxidation level and the computer code model.
7. TW is a sequence where containment heat removal capability is lost.

Table 4.6 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
	SRV Steam Condensation with SP Temperature Exceeding Pool Temperature Limit	SP Cooling or RPV Depressurization
	Downcomer Vent Chugging Load	Drywell Spray to Redistribute Noncondensable 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 Containment Venting

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

Challenge	Mechanism	Strategy
<u>Before Vessel Breach</u>		
1 SP Boundary Load	SRV Discharge with high RPV Noncondensable Gas Content	RPV Depressurization
2 Containment Pressure Load	Mass and Energy Addition (Noncondensable 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 Drywell Spray
<u>At Vessel Breach</u>		
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 & 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 (For above three Mechanisms)	Eliminate Water
3 Pressure Load on RPV Pedestal	Rapid Mass & Energy Addition to the Inside of RPV Pedestal	Early Wetwell Venting to Reduce Initial Pressure
4 Drywell Shell Meltthrough	Thermal Attack of DW Shell by Molten Core Debris	RPV Depressurization
		Drywell Spray or Flooding Reactor Cavity

Table 4.8 Challenges, Mechanisms, and Strategies
During the Late Phase

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

**Table 4.9 Challenges, Mechanisms, and Strategies
During the Release Phase**

Challenge	Mechanism	Strategy
<u>Before or After CF</u> 1 FP in Containment Atmosphere	In-Vessel Release (Core Melt)	SP Scrubbing or Drywell Spray
	Release during VB	RPV Depressurization
	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, Adding Cool Water to SP, or Adding Base to SP
2 Driving Force for FP Release <u>After CF</u> 1 FP Release From Cont. to Outside Containment	Containment Pressure	WW Venting or Small Area Venting with SGTS
	Containment Failure or Venting	WW Venting, Flooding Leak Area, or Containment Flooding
	Isolation Failure	Isolating Failed System, or Same as Above for CF
	Interfacing Systems LOCA (ISL)	Isolating Break Area, RPV Depressurization, or Same as Above for CF
	Pressure and FP Increase in RB	RB Isolation, RB Fire Spray, or Using SGTS
3 FP Release from RB to Environment		

**Table 4.10 Fission Product Groups And Their Inventories
For Peach Bottom**

Group Number	Group Name	Elements in Group	Total Mass (lbm)
1	Xe	Xe + Kr	909
2	I	I + Br	37
3	Cs	Cs + Rb	506
4	Te	Te + Sb, Se	77
5	Sr	Sr	138
6	Ru	Ru + Rh, Pd, Mo, Tc	1285
7	La	La + Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y	1841
8	Ce	Ce + Pu, Np	2182
9	Ba	Ba	231

**Table 4.11 Early And Late Effect Weights Of Source Term Groups
(Reference 54)**

Group	<u>Early Effect Weights</u>			Late Effect Weights ²
	Bone ¹	Lung	Average	
1	0.1	0.05	0.08	0.001
2	1.0	1.0	1.0	0.1
3	0.1	0.09	0.1	1.0
4	0.9	0.7	0.8	0.104
5	0.7	0.9	0.8	0.7
6	0.4	3.8	2.1	1.1
7	2.8	14	8.4	2.6
8	1.6	14	7.8	5.4
9	1.5	0.6	1.1	2.6

¹ 24 Hour bone dose

² Total cancers

Table 4.12 Table Of Doses Due To Release Of 100% Of Core Inventory Of Noble Gases And Equivalent Releases Of Other Groups (Reference 44)

	2 hour decay prior to release	24 hour decay prior to release
95th percentile noble gas dose		
	40 rem	7.5 rem
Fractional release of single groups whose 95th percentile dose is the same as that from 100% noble gas release		
Gases		
Iodine	0.04	0.016
Volatiles		
Cs-Rb	0.065	0.01
Te-Sb	0.05	0.01
Ba-Sr	0.022	0.005
Ru	0.025	0.005
Nonvolatiles		
La	0.001	0.0002
Release Conditions		
3412 MWt PWR end of life core inventory		
2 hour release duration		
Deposition velocity 0.01 m/s		
Miami typical meteorological year data		
Distance (x) 5 miles downwind from reactor		
Equivalent whole body dose		
24 hour exposure to ground contamination		
Lifetime (50 year) commitment for inhalation dose		

Table 4.13 Simplified BWR Fission Product Releases To Containment
For Severe Accident Conditions (Reference 42)

<u>GROUPS</u>	<u>FRCS^a</u>		<u>FCCI</u>	
	<u>H, I</u>	<u>L</u>	<u>Basaltic Concrete</u>	<u>Limestone Concrete</u>
NG	1.0 ^b	1.0	0	0
Cs, I	0.7	0.8	0.15	0.15
Te	0.1	0.15	0.12	0.5
Sr, Ba	6×10^{-3}	6×10^{-3}	0.2	0.7
Ru, Ce, La	3×10^{-5}	3×10^{-5}	6×10^{-3}	0.06
Release Duration	1.5 hrs.		3 hrs. ^c	

^a H, I and L refer to high, intermediate or low RCS pressure, respectively.

^b All entries are fractions of the initial core inventory.

^c Except for Te where the duration of ex-vessel release is extended to 6 hours.

Figure 4.1 Conditional Probability of Accident Progression Bins at Peach Bottom. (Reference 5)

Figure 4.2 Interfaces Between Risk Analysis Steps. (Reference 5)

Figure 4.3 Time Phases of accident progression and Accident Management

Notes

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

Figure 4.4 Early CRET for SBO Sequences

Figure 4.5

Typical Pressure Time History

Figure 4.6

Typical Temperature Time History

Figure 4.7
Progression of Concrete Attack

Figure 4.8
Gas Generation from CCl

Figure 4.9
Corium/Concrete Interface Temperature

Figure 4.10 Drywell Pressure for various Drywell Spray Flowrates
(SBO with spray initiation at 205 minutes)

Figure 4.11 Drywell Temperature for various Drywell Spray Flowrates
(SBO with spray initiation at 205 minutes)

Figure 4.12 Simplified Schematic Of Source Terms
(Reference 5)

Figure 4.13 Time Variation of FP Release to Containment Atmosphere

Figure 4.14 Volumetric Flow Rate for Various Containment Temperatures

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MECHANISMS

STRATEGIES

Figure 4.15 Safety/Objective Tree with Identified Strategies for a Mark I Containment

Figure 4.15 Safety/Objective Tree with Identified Strategies for a Mark I Containment (Continued)

Figure 4.15 Safety/Objective Tree with Identified Strategies for a Mark I Containment (Continued)

Figure 4.15 Safety/Objective Tree with Identified Strategies for a Mark I Containment (Continued)

Figure 4.15 Safety/Objective Tree with Identified Strategies for a Mark I Containment (Continued)

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Figure 4.15 Safety/Objective Tree with Identified Strategies for a Mark I Containment (Continued)

Figure 4.15 Safety/Objective Tree with Identified Strategies for a Mark I Containment (Continued)

Figure 4.15 Safety/Objective Tree with Identified Strategies for a Mark I 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 were discussed in Section 3.3.3. Two important issues determine the availability of instrumentation during a severe accident. The first is the survival of plant instruments under severe accident conditions. Typical instrument qualification pressure and temperature, from Regulatory Guide 1.89 and IEEE 323-1974, are approximately 85 psia and 350 F, respectively (Figures 3.8 and 3.9). The actual environmental conditions in a severe accident may be considerably harsher, particularly if a corium concrete interaction (CCI) has been in progress for some time (see Figure 4.11, for example).

The availability of instruments during station blackout sequences is another important issue, which, as discussed in Section 3.3.3, seems to be plant specific. The only containment parameter that is available in the control room during a station blackout for Peach Bottom is SP temperature, and this is lost after battery depletion. Therefore, it is important to identify other methods to obtain essential parameters under SBO conditions. Section 3.3.3. has discussed this issue and provided some examples.

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.

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 HPSW, and resources such as electric power, pneumatic supply, and water. Section 3.2 provided a detailed discussion of the plant systems and resources that can be used for CRM.

The delivery of water to the containment is required for many strategies presented in Table 5.2. Section 3.2.1 provided a detailed discussion of the alternate water supply systems, the probable sources for additional water, and the special characteristics of different water supply sources. For Peach Bottom, Table 3.3 shows some of the systems that can be used to deliver water to the containment, and Table 3.4 shows some water sources and their capacities.

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 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 Peach Bottom 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

As indicated in Table 5.2, the suppression pool boundary loads from SRV actuation at high RPV pressure, the containment loads associated with HPME, and the fission products released from HPME can all be eliminated or reduced by RPV depressurization. Moreover, the release of the RCS inventory (and the fission products it contains) directly to the outside of the containment in the event of ISL can also be mitigated by RPV depressurization.

RPV depressurization is one of the important actions in the BWR EPGs (Section 3.3.2). Emergency RPV depressurization is called for under the primary containment control guideline when (1) the SP temperature cannot be maintained below HCTL (about 155°F at 1000 psi RPV pressure for Peach Bottom), (2) the drywell temperature cannot be maintained below the drywell design temperature limit or ADS qualification temperature, whichever is smaller, (3) the containment pressure cannot be maintained below the pressure suppression pressure (PSP, 40 psig for Peach Bottom), (4) the SP water level cannot be maintained above the heat capacity level limit (HCLL, a level adjustment for HCTL) 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 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 hold the SRVs open. 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 discussed in Section 3.2.4.2.

As a result of the CPI program, the NRC staff has recommended to the commissioners an enhanced RPV depressurization system for Mark I containments [46]. The recommended ADS enhancements 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 noncondensable gases is generated in the RPV from cladding oxidation, (2) to avoid the challenges associated with HPME, and (3) to reduce the amount of FPs released to outside the containment during an ISL event. 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 reactor building or the environment and a relatively low temperature and radiation level in the primary containment.

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

5.2.3 Strategies Related to Containment Venting

Containment venting is recommended in the BWR EPGs as a means to prevent containment failure due to high pressure. The BWR EPGs provides EOP guidance for the operator to carry out containment venting before the pressure reaches the primary containment pressure limit (PCPL).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. These venting strategies are discussed in the following.

Containment Venting for Containment Pressure Control: 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 was discussed in detail in Section 3.2.4. It is the strategy that can prevent containment failure and subsequent loss of core injection for cases where CHR is lost or insufficient. It is also the only action that plant personnel can take to prevent a containment pressure failure due to noncondensable 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 3.2.4.2.

The determination of the venting pressure, PCPL, is another important issue discussed in Section 3.2.4.2. 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 [33]. The pressure rise rate in the containment during an accident is directly related to the mass and energy input rate and provides an indication of the time and vent area required to achieve pressure control (Section 3.2.4.2).

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.

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 (Table 3.7). 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. Additional discussion can be found in Section 3.3.3.

Early Venting to Prevent a Containment Pressure Failure: 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 inevitable even with reduced initial containment pressure, such as might occur due to direct containment heating (DCH) or an early drywell shell melt-through (DSM), 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 4.2.3.3 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 system can be used (Section 4.2.1.2.1, Hydrogen Combustion). 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 some 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 (Section 4.2.3.2). 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 reactor building 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 reactor building to determine the approximate location of the leak. However, the instruments in the secondary containment (reactor building) 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 [34], 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 more vent paths 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.

Potential Adverse Effects: As discussed in Section 3.2.4.2, the important adverse effects include (1) loss of plant safety equipment due to containment depressurization and SP flashing, (2) reactor building contamination and resultant loss of function of safety related equipment or loss of accessibility to the reactor building, and (3) fission product release to the environment. Methods to avoid or mitigate these adverse effects are also discussed in Section 3.2.4.2. The use of a hardened vent path, as recommended by the CPI program, will further lessen the concern of RB contamination.

Operator Action Concerns: 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 the Peach Bottom TRIP, which requires the operator to close the vent path at a pressure of 10 psi below the venting pressure.

Containment venting requires operator decisions and actions. In Peach Bottom the time required to successfully carry out the venting procedure is estimated to be about 30 minutes for the ATWS sequences, where power is available but many vent paths must be opened, and about 2 hours for the station blackout (SBO) sequences where power is unavailable and the vent valves have to be opened manually [25]. The environmental conditions such as heat and radiation level may be too high for the operators to manually open the valves, although this is highly plant dependent. For Peach Bottom, a procedure review and a human reliability analysis estimated that the success probability of venting is 0.7 for ATWS but zero for SBO sequences because of the hazardous environment at the vent path valve location in this plant [25]. A provision for dc operation of the isolation valves, as recommended by the CPI program, will certainly increase the success probability of venting for SBO sequences.

In accordance with the requirements of NUREG-0696 functions (Table 3.6), 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 a contaminated containment atmosphere which will often exist when venting could be useful. Responsibility on venting decision should be clearly defined and explicit and unambiguous guidance should be given to the operators.

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 shell melt-through, (Section 3.2.3). Furthermore, drywell spray also has a potential to mitigate the effects of hydrogen combustion (Section 4.2.1.2.1), to reduce the challenges associated with HPME (Section 4.2.1.2.2), and to prevent excessive SP boundary loads due to chugging (Section 4.2.1.1).

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, 17 psig for Peach Bottom). Drywell spray is also called for in the BWR EPGs when containment hydrogen and oxygen concentrations cannot be controlled to below predefined limits (6% for hydrogen and 5% for oxygen) 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 is the topic of the next strategy and 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.

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 -2 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 (Section 3.3.2.1).

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 4.2.1.2.2). 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.

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.

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 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 (Sectin 4.2.1.2.2).

Finally, the use of an alternate source with untreated water may clog the containment spray system.

Operator Action Concerns: In using drywell spray to meet a variety of challenges occurring at different phases of an accident with very different containment conditions it is necessary to be aware of 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 and the Peach Bottom TRIP. However, modification to existing guidelines or additional guidelines may be required because of the extended applications of drywell spray in CRM.³

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 posed by DCH and DSM. 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 and (b) provide an

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 water supply system is available. One solution to this problem is to modify the system such that it 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 noncondensable gases in the drywell.

overlying water pool for fission product scrubbing. The effects of water on DCH, DSM, and CCI still involve some uncertainties, but are expected to be beneficial and desirable.

Most of the time, only indirect inferences are available to deduce the existence of the challenges that are addressed by this strategy (Tables 5.1 and 5.2). Furthermore, to assure sufficient water in the reactor cavity before vessel breach, the drywell spray needs to be initiated considerably before vessel breach.

Operator Actions and Equipment Requirements: Drywell spray is the only means to add water to the reactor cavity before vessel breach. 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 (Sections 4.2.3.1 and 4.2.3.2 on revolatilization from RCS), adding water via the drywell spray can provide an additional fission product scrubbing capability and also cool the containment atmosphere.

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 a steam explosion at vessel breach (Section 4.2.1.2.2). This pressure increase, 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 expected beneficial effects for the mitigation of DCH and DSM. 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.

Operator Action Concerns: 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.

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 4.2.1.2.1 and 4.2.2). 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 4.2.1.2.1.)

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.

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., CADS and CACS, need to be observed to avoid damage to these systems (Section 3.2.4.1). 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 (both are discussed in Section 4.2.3.2).

Reactor building 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.

As shown in Figure 4.14, 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.

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 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 4.2.2.

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 the drywell vent paths are not flooded and remain operational for the duration of the accident.

As discussed in Section 4.2.2, 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 Reactor Building Fission Product Retention

In the event of primary containment failure fission products are usually discharged to the reactor building before they pass into the environment. Therefore the reactor building provides the last opportunity to mitigate the release of radioactive materials to the environment. Retention of fission products in the reactor building can be achieved by (1) natural deposition on reactor building structures, (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 4.2.3.2. Additional discussion is provided below.

Using the SGTS to Reduce Fission Product Release: The BWR EPGs call for the reactor building Heating, Ventilation, and Air-conditioning (HVAC) system to be isolated and the SGTS to be initiated when the reactor building HVAC exhaust radiation level exceeds its isolation setpoint. The SGTS is used to remove fission products from the reactor building 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 4.2.3.2). 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 reactor building is not designed to withstand a significant pressure load and substantial leakage will develop in the RB 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 reactor building. This adverse effect becomes important if the filters of the SGTS have failed and the RB pressure is low (i.e., a low containment leak rate).

Additional issues regarding the SGTS were discussed in Section 4.2.3.2.

Reactor Building Fire Spray for Fission Product Retention:
The reactor building 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 (Section 3.2.3).

The fire system for Peach Bottom has been discussed in Section 3.1.3.3. The capacity, the head, the water and power sources of the fire system for Peach Bottom are presented in Table 3.3. Remote manual operation of the fire system, if available, may need electric power. Since the diesel driven part of the fire spray system does not depend on plant electric power, the ability to manually operate the system makes this strategy available during SBO sequences.

The need for fission product removal from the reactor building atmosphere can be inferred from high offsite radioactivity and high reactor building area radiation readings. However, the reactor building 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.

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

5.2.10 Other Strategies

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

Table 5.1 Challenges and the Parameters to Identify Challenges

		Parameters	
	Challenge	Direct	Indirect
1	Loss of Resources	Instrument Indication for Electric Power, Water & Pneumatic Supply	Functionality of Plant Systems
2	SP Dynamic Loads		RPV Pressure, Containment Pressure, and SP Temp. & Water Level
3	Containment Pressure	Containment Pressure	
4	Drywell Temperature	Drywell Temperature	
5	Drywell Shell Meltthrough		RPV Breach Condition, Reactor Cavity Condition
6	Rapid Containment Pressurization Related to HPME		RPV and Cont. Conditions
7	Burn of Combustible Gases	Cont. Gas Concentration	
8	Core-Concrete Interaction (CCI)		Cont. Pres., Temp, Gas Conc., and FP (Radiation)
9	FP in Containment Atmosphere	Containment Radiation	
10	FP From Containment to RB	RB or Env. Radiation	Containment Radiation
11	FP From RCS to RB	RB and Env. Radiation	Containment Radiation
12	FP From RB to Environment	RB and Env. Radiation	

Table 5.2 Strategies and the Challenges Addressed by the Strategies

Challenge (See Table 3.1 for Description)													
	Strategy	Loss of Re-sources	SP Dynamic Loads	Containment Pressure	Drywell Temperature	Drywell Shell Melt-through	Rapid Containment Pressurization Related to HPME	Burn of Combustible Gases	Core-Concrete Interaction (CCI)	FP in Containment Atmosphere	FP From Containment to RB	FP From RCS to RB	FP From RB to Environment
1	Resource Management	X											
2	RPV Depressurization		X				X			X		X	
3	Netwell Venting			X			X	X			X		
4	Drywell Spray		X	X	X	X	X	X	X	X			
5	Reactor Cavity Flooding												
6	Combustible Gas Control							X			X		
7	Leak Area Flooding										X		
8	Primary Containment Flooding								X	X	X		
9	RB FP Retention												X
10	Other Strategies												
	Containment Cooling			X	X					X			
	SP Cooling	X	X							X			
	SP Water Level Control	X	X										
	Isolate Leak Area										X	X	

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 Peach Bottom Atomic Power Station (PBAPS) was used as the surrogate plant for this assessment. The PBAPS Transient Response Implementation Plan (TRIP), instead of the BWR EPGs, was 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 provide examples of 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, LOCA, 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. These sequences are selected to provide examples for strategy implementation and assessment only. No attempt was made to include all sequences each license has to consider.

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 all off-site and on-site ac power. This leads in a Mark I BWR to the loss of all active engineered safety features except the steam powered High Pressure Core Injection (HPCI) system and Reactor Core Isolation Cooling (RCIC) system. Since both high pressure core

cooling systems require dc power for control, they may fail after the depletion of station batteries. 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 sequence described is normally termed a slow SBO sequence, as opposed to a fast SBO sequence, where dc power is also lost at the beginning of the sequence.

6.2.1 Characteristics of SBO Sequences

SBO sequences are characterized by the loss of most of plant instruments and equipment. As discussed in Section 3.3.3, for Peach Bottom the only containment indication available after the loss of ac power is SP temperature and this is lost after dc power is depleted. 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, and in Section 3.2.1 for primary containment cooling and water supply.

6.2.2 Containment Response to SBO Sequences

Figures 6.1 and 6.2 present containment pressure and temperature time histories of a slow SBO sequence for Peach Bottom (TBI case of NUREG/CR-4624 Volume 1 [16]). These histories were obtained with the STCP code. In the TBI Case, the plant batteries are assumed to be depleted six hours after accident initiation and recovery and mitigating actions are assumed to be unsuccessful. As a result, the core starts to melt at about 11 hours, and the RPV is breached at about 12 hours after accident initiation. Since control of the SRV is lost after the loss of dc power, the RPV pressure rises again after battery depletion and the vessel is breached at high pressure. This results in a pressure spike in the containment at the time of vessel blowdown. After the initial spike the pressure drops immediately due to energy absorption by containment structures. This is followed by a continuous pressure rise, caused by the energy and noncondensable gases generated from CCI. The containment pressure increases continuously until the failure pressure of 132 psia is reached at about 15 hours after accident initiation. The containment temperature basically follows the same trend as containment pressure before containment failure, but continues to rise even after containment failure, reaching 1000°F a few hours after failure.

In a fast SBO sequence, core melt starts early because all core injection is lost at accident initiation. Figures 6.3 and 6.4 show the containment pressure and temperature time histories for this case [39]. To compare the effect of different battery lives,

Figures 6.5 and 6.6 show the containment pressure and temperature responses obtained from STCP calculations for a battery life of zero (fast SBO), 6, and 12 hours. Containment pressures and temperatures of the various cases exhibit similar behavior after battery depletion. The increase in the containment pressure and temperature at the time of battery depletion, as battery life is extended, is due to the additional decay heat released during the longer time period. In the extreme case, when plant dc power, and thus high pressure 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.

As previously discussed, significant uncertainties exist in the understanding of some severe accident phenomena, and in the ability to predict containment responses accurately. However, the results of the STCP 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 slow 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 fast 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 [48]). 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 (Table 3.6).

Tables 6.1 and 6.2 show the plant status indications and the corresponding EPG actions. There are no significant containment challenges during the early phases of the accident. Steam generated in the RPV from decay heat is discharged through the SRVs to the SP and causes a temperature increase in the SP, which, in turn, causes a slow pressure and temperature increase in the containment atmosphere. As noted in Table 6.1 containment actions

called for by the EPGs are not likely to be carried out because of the lack of instrument readings or unavailability of required equipment.

As shown in Tables 6.1 and 6.2, core melt starts at about 5 hours after battery depletion for the slow SBO case and about 2 hours after accident initiation for the fast 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 pressure and temperature continue to increase and there is a significant pressure jump after the core slump (Figures 6.1 and 6.3). The challenges associated with these conditions are not significant and actions are unlikely because of the lack of electric power. The most important challenges to containment integrity during this time are those associated with HPME, occurring at the end of this phase. Since the STCP does not have a DCH model, the pressure rise due to HPME shown in Figure 6.5, 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.

In some cases early wetwell venting could be used to reduce the base pressure and thus the impact of HPME (Section 5.2.3). However, according to STCP calculations, this may not be practical for the present case because containment pressure before core melt is low (only about 12 psig for the slow SBO case and about 2 psig for the fast SBO case). MELCOR calculations for a similar case show a somewhat higher pressure increase before core melt. Containment pressure before vessel breach, as shown in Figures 6.1 and 6.3, reaches its highest value shortly after the core slump, and venting during this time period would give the greatest base pressure reduction. However, it is impossible for the operators to monitor core melt progression in enough detail to use this narrow time window between the core slump and vessel breach for venting. Early venting might be a more attractive option if containment pressure is higher before core melt as would be expected if high pressure injection can be maintained for a longer time. The ability to vent the containment during an SBO sequence has been discussed in Section 3.2.4.2.

The probability of early containment failure from DSM can be reduced by flooding the reactor cavity before vessel breach (Section 5.2.5). The capacity of the reactor cavity sump for Peach Bottom is approximately 2,000 gallons and approximately 12,000 gallons of water are required to flood the drywell floor to the bottom of the main vents, about 18 inches above the drywell floor (cross sectional area of about 100 square meters). The time required to flood this area depends on the flow rate and is not expected to be significant. For example, it takes only a few minutes for the fire water system, with a capacity of 2,500 gpm, to flood the drywell floor. The key question is the availability of water. Water, if available, 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 to deliver water to the containment for reactor cavity flooding. The drywell spray, once started, will also provide a heat sink for the energy released during HPME (Section 5.2.4).

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 EPG limits (e.g., 40 psig and 281° F), 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 noncondensable gases resulting from CCI (Section 4.2.2). As shown in Table 6.1 and 6.2, the containment venting pressure (PCPL, 100 psig for Peach Bottom) is reached 14 hours after accident initiation for the slow SBO sequence and about 5 hours after accident initiation for the fast SBO sequence.

Figures 6.7 and 6.8 show the drywell pressure and temperature response with drywell sprays at various flow rates for a fast SBO sequence. Both drywell pressure and temperature decrease with increasing spray flow rate¹. The effect of the spray on drywell temperature is particularly important. Even a moderate flow e.g. 500 gpm, can maintain the drywell temperature below 400° F. While this still exceeds the containment design and equipment qualification values, it is not expected to pose a serious challenge to containment integrity or to equipment and instruments. The pressure response as shown in Figure 6.7 is highly influenced by the flow rate. For a failure pressure of 150 psia, a spray flow rate of 6,000 gpm would delay failure by more than three hours (from about 350 minutes to 550 minutes in Figure 6.7). Since the significant ex-vessel FP release occurs when the CCI is most vigorous (at about 350 minutes in Figure 6.7), this delay of

¹The drywell pressure and temperature responses for spray flow rates from 500 to 6000 gpm are presented in Figures 6.7 and 6.8. Only lower flow rates are available (e.g., 2500 gpm or less from fire protection system, Table 3.3) if ac power is not recovered. The containment free volume will be reduced if the spray is not in a recirculation mode. With a 2500 gpm spray, the containment free volume will be reduced by 7% of its original volume every hour.

containment failure would have a significant impact on radioactivity release. However, it must be pointed out again that significant uncertainty exist in the ability to predict accident progression accurately and the results of Figure 6.7 are used as an example to show the potential beneficial effect of removing energy from the containment atmosphere. (As discussed in Section 3.2.3, drywell spray, as modeled in the CORCON module of STCP, does not have a significant effect on mitigating the progression of CCI, and the primary reason for the pressure reduction shown in Figure 6.7 is due to energy removal from the containment atmosphere by the spray.)

Figures 6.9 and 6.10 show the containment pressure and temperature response with containment venting at various venting areas. The pressure and temperature increase after vessel breach is primarily due to the hot gases released from CCI (Section 4.2.2). As shown in Figure 6.9, the equivalent orifice area required to maintain containment pressure below the venting initiation pressure is not large. However, Figure 6.10 shows a faster containment temperature increases as the venting area increases. 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.

As discussed in Section 4.2.2, in this phase of the accident CCI is the source of many containment loads. Flooding the reactor cavity before vessel breach and continuously adding water to the corium increases the probability of slowing down the progress of CCI, eventually terminating it. The strategies of reactor cavity and containment flooding are important for this phase as discussed in Sections 5.2.5 and 5.2.8.

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 slow SBO sequence and in Table 6.2 for a fast SBO sequence. Challenges and strategies are the same for both sequences and have been discussed in detail in Section 4.2.3. Individual strategies have been discussed in Section 5, and an assessment of the effectiveness of plant systems to mitigate FP release, e.g., drywell 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 early (within about 3 hours of vessel breach, Section 3.1.1.2) in the drywell, releasing radionuclides without going through the SP. This scenario will be used as the base case to discuss the effects of a few strategies on FP release. As indicated in Section 4.1.1, the objective of such

strategies is to change the release condition from one represented by an APB with a greater FP release potential to one with less. Figure 6.11 shows a partial release CRET which can address the effects of a few strategies where FP release data are available from NUREG-1150 APET analysis (NUREG/CR-4551, [54]). For example, Figure 6.11 shows that the use of wetwell venting would change the release category from APB 45 to APB 37.

Table 6.3.a shows the release fractions of the nine radionuclide groups for the various release categories presented in Figure 6.11. Table 6.3.b shows the equivalent fraction of a single radionuclide, which can be used to represent the combined effect of all nine groups presented in Table 6.3.a (See Section 4.2.3.1 for additional discussion). For example, Table 6.3.b shows that the early effect of the base case release is equivalent to the release of 1.723 times the original iodine inventory. The late effect of the release is equivalent to 1.558 times the original inventory of cesium.

As shown in Table 6.3, wetwell venting before vessel breach or during significant CCI activity would reduce the effective release by more than an order of magnitude. In this case (APB 37) the reactor building is assumed to be bypassed, either due to the use of a hard vent path or due to a breach of the RB. APB 29 of Table 6.3 reflects the effect of a late drywell spray as a result of a late power recovery, and APB 43 shows the effect of the reactor building fire spray. The early drywell leak case, APB 30, includes the bypass of the reactor building. Leaks directly into the refueling bay of the reactor building, such as those that occur in a drywell head leak case, would result in a FP release similar to that of APB 30. As shown in Table 6.3, the release is comparable to that of the base case and is significant. This shows the importance of flooding the leak area if drywell spray is not available.

As also shown in Table 6.3, a late FP release, long after the important FP release sources have subsided, would release significantly less FPs to the environment. Late containment venting (APB 39) can be used to preserve containment integrity, but the reduction in FP release achieved by the late venting is not significant when compared with that from late containment failure (APB 50). A late drywell leak would result in the smallest FP release of all cases presented in Table 6.3.

6.3 ATWS Sequences

The ATWS sequences discussed in this section are those initiated by an MSIV closure at full power. Reactor power, after a successful automatic recirculation pump trip and RPV water level control, would still exceed the containment heat removal (CHR) capability of the RHR system. The discharge of the RPV steam to the 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, and core melt, then vessel breach, would follow. This sequence is commonly referred as a slow ATWS sequence.

In a fast ATWS sequence, core injection is lost before containment failure, and, as a result, 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 3.2.4.2.

6.3.2 Slow ATWS Sequences

6.3.2.1 Containment Responses of a Slow ATWS Sequence

Figures 6.12 and 6.13 present the containment pressure and temperature time histories of a slow ATWS sequence calculated by the STCP code for Peach Bottom (TC1 case of NUREG/CR-4624 [16]). In the calculation, the reactor is assumed to be at 21% rated thermal power. The containment pressure, as shown in Figure 6.12, reaches its failure point (117 psig) at about 90 minutes after accident initiation. The ECCS fails after containment failure, and core melt starts at about 135 minutes. Vessel breach occurs at 230 minutes after accident initiation. Since a catastrophic containment failure (a failure area of 7 ft²) is assumed in the calculation, containment pressure drops to atmospheric level and remains there after the containment fails. Containment temperature, as shown in Figure 6.13, reaches its design value early in the accident, drops slightly after containment failure, and then rises to over 1000°F a few hours after vessel breach.

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 TC1 sequence, shown in Figures 6.12 and 6.13, 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 an hour 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. RPV depressurization is assumed to be successful and the low pressure ECCS system is 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.

Containment venting pressure, PCPL, is reached at about 75 minutes after accident initiation. The effect of containment venting on containment pressure and temperature are shown in Figures 6.14 and 6.15, which are calculated with STCP for various vent areas. It is assumed in these calculations that the core power is controlled at 15%, which is between the 21% used in the TC1 calculation shown in Figures 6.12 and 6.13 and the 12.7% predicted by the BWR-LTAS computer code [25]. As shown in Figure 6.14, a vent area of about 2.5 ft² is needed to control the containment pressure at about 100 psia. This requires the opening of at least two 18-in vent paths. In fact, a detailed choked flow calculation, using the RELAP5 thermal-hydraulic code which considers the loss in the actual vent lines, shows that the equivalent diameters for the four 18-in lines, i.e., wetwell

exhaust, wetwell supply, drywell exhaust, drywell supply, are 4.3, 13.4, 15.6, 4.0 inches, respectively [25]. As a result, all four vent paths are required to provide sufficient vent area for containment pressure control.

In Peach Bottom, the vent paths are opened in the order shown in Table 3.5. For an ATWS case, where the containment pressurization rate is greater than 30 psi/hr, the 6-in lines are bypassed because considerable operator actions outside the control room are required to prepare them for venting (Section 3.2.4.1). The time needed to open the required vent paths has been estimated, by a review and analysis of the Peach Bottom procedures and a task analysis of the human actions required to carry out the procedure, to be about 30 minutes [25]. Venting actions should therefore be initiated about 30 minutes before PCPL is reached, or about 45 minutes after accident initiation for the TC1 case. The Peach Bottom TRIP requires the operator to prepare to vent the containment when containment pressure reaches 17 psig, and, as shown in Figure 6.12, there seems to be sufficient time for the operator to open the required vent paths before PCPL is exceeded. 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 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 5.2.2).

Figures 6.16 and 6.17 show the partial pressure of the noncondensable gases in the drywell and the wetwell, for the cases presented in Figures 6.14 and 6.15. At the beginning of the accident, RPV steam is discharged to the SP, causing the pool temperature to rise and the vapor pressure in the wetwell to increase. The pressure increase in the wetwell will then cause a flow from the wetwell to the drywell through the vacuum breakers. Noncondensable gases originally in the wetwell will be carried to the drywell and this results in a decrease of noncondensable gas partial pressure in the wetwell and a corresponding increase in the drywell. After the opening of the vent paths, the noncondensable gases originally in the containment (both the drywell and the wetwell) are discharged to the outside and replaced by the steam released from the SP. Shortly after containment venting, the containment atmosphere would be practically void of noncondensable gases and become full of steam. The use of a high capacity, cold containment spray in this steam environment could cause a rapid containment pressure decrease and an unacceptable negative pressure load, and should therefore be avoided. The CADS or CACS in the

containment vent and purge mode (Sections 3.2.4.1 and 5.2.6) 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 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 Responses of a Fast ATWS Sequence

Figures 6.18 and 6.19 present the containment pressure and temperature time histories, respectively, of a fast ATWS sequence for Peach Bottom (TC2 case of NUREG/CR-4624 [16]). The difference between this case and the slow ATWS case is that in the present case RPV depressurization is not successful. Core injection is lost after the failure of the HPCI system due to high SP temperature (200°F) and the failure of the RCIC due to high turbine exhaust back pressure (25 psia containment pressure)[16].

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 decreases slightly 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, after which the containment pressure increases considerably (about 50 psi, Figure 6.18). The high pressure vessel blowdown that follows results in a significant pressure spike in the containment and containment failure. Containment temperature basically follows the same trend as containment pressure before containment failure, but containment temperature continues to rise even after containment failure and could reach 1000°F a few hours after failure.

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 TC2 sequence, shown in Figures 6.18 and 6.19, are used to provide timing and containment conditions for Table 6.5.

The challenges to containment integrity for the TC2 sequence are similar to those for a fast SBO sequence (comparing Figures 6.3 and 6.18). The major difference between these two cases is that the base containment pressure before vessel breach for the TC1 case is higher than that for the slow SBO case (80 psi vs 40 psi). For the TC2 case, the pressure rise before core melt is higher because of higher reactor power, and the pressure rise during core degradation is higher because of a higher zirconium oxidation level (about 60% for TC2 and 30% for fast SBO). The high base pressure (the pressure immediately before vessel breach) increases the probability of containment failure at vessel breach. While the failure pressure (assumed to be 117 psig) is not reached during vessel blowdown for the SBO case (Figure 6.3), it is exceeded in the TC2 case (Figure 6.18). It should be noted again, however, that there are considerable uncertainties in both the containment pressure capability (Section 3.1.1.1) and the ability to accurately predict accident progression.

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 6.18, the venting pressure is reached during vessel blowdown. Assuming that sufficient vent paths can be opened at the time venting pressure (assumed to be 100 psig) is reached, Figure 6.20 shows the pressure responses for various venting areas. The calculated peak containment pressure is not reduced significantly for the venting cases shown in Figure 6.20, and its value exceeds the assumed containment failure pressure (117 psig) for all cases. However, the failure pressure is exceeded only for a very short time duration. If a sufficient venting area could be opened in time, the containment pressure load would be limited to a short impulse load, not expected to cause structural damage. However, because of the short time scale of HPME, it is unlikely a sufficient venting area can be opened in time.

If the accident proceeds as predicted in Figure 6.18, the only practical means to prevent early containment failure is to carry out an early wetwell venting to reduce the base containment pressure before vessel breach. The application and adverse effects of early wetwell venting have been discussed in Section 5.2.3.

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 (Drywell temperature is only about 200°F, Figure 6.19).

6.4 TW Sequences

Accidents involving the loss of long term containment heat removal (CHR) (usually designated as TW sequences) 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 TW 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.

Figures 6.21 and 6.22 present containment pressure and temperature time histories of a TW sequence (TW case of BMI-2104 [15]) for Peach Bottom. The pressure in the containment increases slowly, but monotonically, as reactor decay heat is discharged to the containment, until the failure pressure is reached more than 30 hours after accident inception. Wetwell temperature basically follows the saturation temperature of containment pressure and remains moderate (less than 350 °F) up to the time of containment failure. According to STCP, drywell temperature, as shown in Figure 6.22, lags behind wetwell temperature and is not expected to affect the performance of the instruments and equipment in the drywell.

The operator actions required to mitigate the various challenges of a TW sequence are similar to those of a slow ATWS sequence (Table 6.1), but with a much longer time scale. The most important operator action, containment venting, is not required until more than 20 hours after accident inception. The probability of successful containment venting is very high because of the long time available for the operator to prepare. This reduces the significance of the TW sequences, and as a result, this sequence does not contribute significantly to the total core damage frequency of Peach Bottom in NUREG-1150 [5].

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

6.5 AE Sequences

One of the important plant damage states considered in NUREG 1150 for Peach Bottom is a large or medium LOCA followed by failure of all core injection. As discussed in the NUREG-1150 study, a likely scenario for the initiation of this accident is as follows: the high pressure injection fails because of low RPV steam pressure, and the low pressure core cooling systems fail to actuate primarily because of miscalibration faults in the pressure sensors [7].

Containment Responses of an AE Sequence: The containment pressure and temperature for a large break LOCA with loss of all injection are shown in Figures 6.23 and 6.24, [15], for Peach Bottom. The containment pressure and temperature rises rapidly to about 40 psia and 250°F, immediately after the break of the primary system. After the initial blowdown, the containment pressure drops to about 30 psia due to energy absorption by containment structures. Containment pressure stays at about 30 psia during the course of core melt (starting at about 11 minutes) until the core slumps at about 30 minutes. Containment pressure rises sharply after core slump and the failure pressure is reached shortly after core slump (less than 10 minutes after slump). The containment temperature, as shown in Figure 6.24, generally follows the same trend as containment pressure. It increases sharply after the core slumps and reaches more than 2,200°F just before containment failure.

Special Characteristics of an AE sequence: In an AE sequence, the in-vessel release, before vessel bottom head failure (defined here as vessel breach), goes directly to the drywell. The most significant challenge to containment integrity, as shown in Figure 6.23, is the containment pressurization after the core slumps but before the vessel breaches. The release of the high temperature noncondensable gases (primarily hydrogen from zirconium oxidation) directly to the drywell atmosphere results in an overpressure failure. This pressure increase can be significantly reduced if the drywell temperature can be controlled to a value much lower than that shown in Figure 6.24. For example, containment pressure can be reduced by a factor of two to three if the containment temperature can be controlled to below 500°F. (The effect of temperature reduction on containment pressure has been discussed in Section 4.2.2.). Since drywell spray is very effective in reducing drywell temperature (Figure 6.8), it should be used early in the accident (within 30 minutes of accident initiation) to control containment temperature and pressure.

Challenges and Strategies for an AE Sequence: An Alert will be declared at the beginning of the accident because the primary coolant leak rate is greater than 50 gpm, and a Site Emergency may also be declared early because of the loss of core injection in

addition to the primary system break. As a result, both the TSC and EOF would be activated early.

Containment control guidelines will be entered early in the accident because of the containment conditions immediately after vessel blowdown. The containment pressure before core degradation is about 30 psia (Figure 6.23) and the containment temperature at this time is about 200°F (Figure 6.24). These values are below the conditions for drywell spray actuation in the Peach Bottom procedures. Drywell spray conditions, as shown in Figures 6.23 and 6.24, will be reached during the time when the containment pressure and temperature rise rapidly. Less than 10 minutes elapse between the time the spray initiation condition is reached and the time the containment fails. It is not clear whether sufficient time exists to implement a drywell spray with sufficient flow rate to control the containment temperature and pressure effectively, particularly if the normal spray source is not available and an alternate water source is needed. As discussed above, the lack of drywell spray would have a significant effect on the pressure and temperature responses, and it is important to ensure that drywell spray is on before a large amount of high temperature gases are released to the containment atmosphere. As shown in Figure 6.23, this release occurs within 30 minutes after accident initiation.

It should be noted that the predicted containment pressure and temperature responses depend on the core slumping scenario assumed. The pressure spike of over 100 psi predicted in Figure 6.23 is treated as a pessimistic estimate in the NUREG-1150 analysis [9], where a pressure rise of about 45 psi is considered as the more likely central estimate. Containment failure is therefore not likely before vessel breach even without the actuation of the drywell spray.

If the containment does not fail before vessel breach, the challenges and the strategies required to meet them would be similar to those of the SBO sequences already discussed in Section 6.2 (Tables 6.1 and 6.2). Since the in-vessel fission product release goes directly to the drywell, bypassing the scrubbing of the SP, the use of the drywell spray becomes more important than that in the SBO sequences.

6.6 V Sequences

V sequences involve the breach of the reactor coolant system (RCS) pressure boundary at an interface with a low pressure system (Section 4.2.3.2). 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 reactor building. The V sequence is not included in the plant damage states (PDS) considered in NUREG-1150 [5] due to the low core damage frequency (CDF) associated with this type of sequence.

However, since the release bypasses the primary containment and the suppression pool, it is a high risk sequence and will be discussed in the following.

Figures 6.25 and 6.26 present the containment pressure and temperature responses of a sample V sequence [16], which assumes a break in the low pressure ECCS system in the reactor building and the loss of all core injection. Since the break is outside the primary containment there is no appreciable pressure and temperature increases in the containment before vessel breach. The response in the RPV will be similar to that occurring in a LOCA sequence with the loss of all core injection. 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.

For the calculation of Figures 6.25 and 6.26, the break area is assumed to be a 6-inch diameter rupture. If recovery actions, i.e., leak area isolation or core injection, are not successful, core melt would start in about 30 minutes, and vessel breach would occur at about 100 minutes after accident initiation. After vessel breach, containment pressure would rise to about 22 psia and then decrease to atmospheric pressure in approximately 60 minutes. Containment pressure would rise again at 160 minutes to above 50 psia due to the gas and energy generated from CCI, and decreases again to slightly above atmospheric pressure at about 350 minutes as CCI slows down. Drywell temperature, as shown in Figure 6.26, continues to increase during the accident and reaches about 700°F at the end of the calculation.

The blowdown of the high temperature steam from the RPV directly to the reactor building, bypassing the primary containment, will result in a high temperature atmosphere in the area near the break and an entry condition for the secondary containment control procedure. (The setpoint for the secondary containment area temperature alarm for Peach Bottom depends on the temperature sensor location and varies from 105°F to 165°F [24]. The temperature in the reactor building is above 200°F in Figure 6.26.) 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 reactor building conditions, the operator is then instructed by the procedures to shutdown the reactor, enter RPV control guidelines, and perform emergency RPV depressurization.

The reactor building 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 reactor building area temperatures may also be caused by a fire in that particular area. However, with the instrumentation indication available for the RPV and for the area radiation

monitoring system, it should not be difficult to distinguish one from the other.

Isolation of the system that leaks to the outside of the primary containment could eliminate the source of the accident and terminate its progression. Alternately, 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 is predicted to vary from 200 to 1000 psi before vessel breach. RPV depressurization will therefore be required and will be effective in reducing the driving force during this phase. Via 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 procedures in the EPGs seem to address release control as discussed above.

The release could also be reduced by flooding the pipe that leads to the leak area or keep the leak area submerged under water (Section 4.2.3.2). 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 (up to 50 psia, from 150 to 300 minutes, Figure 6.26) 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 reactor building 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 is desirable to open all available wetwell vent paths as early as possible to have the maximum amount of the release passing 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).

6.7 Insights

The assessment presented in this section is based in large part on the calculational results obtained from the STCP. Although important features of accident progression can be identified from the STCP results, actual values of containment loading and the timing of this loading may be quite different from those predicted by the code, since significant uncertainties exist in our understanding of some severe accident phenomena modeled in the code. Clearly therefore, predicted containment loading is somewhat code dependent. For instance, MELCOR predicts higher DW temperature than STCP before vessel breach for most transient scenarios. This may in part be due to how heat transfer from the RPV surface to the DW atmosphere is modeled.

The pressure capability of the containment is another important SAM issue that has significant uncertainty. A further complication is the containment strength's temperature dependence, which is important in sequences where high temperature gases are released to the containment atmosphere or where significant heat transfer from the hot core debris takes place, during CCI. It should be noted that the NUREG-1150 estimate of containment pressure capability is based on an "as-built" containment, and does not account for corrosion or other possible deterioration over the life of the plant. The containment pressure capability used in the NUREG-1150 analyses for Peach Bottom is presented in Figure 3.2 for three temperatures. Figure 6.27 shows an estimate of the pressure capability as a linear function of temperature using the data presented in Figure 3.2.

The importance of containment temperature on containment failure can be illustrated by the STCP calculated containment pressure and temperature responses for the containment venting cases presented in Figures 6.9 and 6.10. As shown in these figures, containment venting can prevent the containment pressure from reaching a predefined value, ie., PCPL, but, the containment temperature will continue to rise (at a faster rise rate with larger venting area) and the containment can fail due to a combination of pressure and temperature loads. Figure 6.28 shows the containment pressure responses and the containment pressure capabilities (based on the temperature responses in Figure 6.10 and the temperature dependence in Figure 6.27) for two of the cases presented in Figure 6.9. Although the containment pressure decreases after containment venting begins (when containment pressure reaches 115 psia), containment pressure capability, which decreases as containment temperature increases (using the median

value in Figure 6.27), can be exceeded shortly after containment venting. Additional removal of the energy from the containment atmosphere is therefore needed to prevent containment failure in the venting case. Such removal can be achieved by the operation of the containment cooling system or the operation of the drywell spray.

Figure 6.29 shows the containment pressure responses and the containment pressure capabilities for two of the (spray) cases presented in Figures 6.7 and 6.8. Drywell spray, as shown in Figure 6.29, can reduce containment pressure by removing energy from the atmosphere and thus delay containment failure by several hours. It also keeps the containment temperature low and, as consequence, maintains the containment pressure capability high. However, containment failure could still occur due to the buildup of noncondensable gases in the containment. As a result, both containment venting and containment spray are likely to be needed to prevent containment failure for cases where high temperature noncondensable gases are released to the containment atmosphere. Figures 6.30 and 6.31 show the containment pressure and temperature responses, respectively, for a case with both drywell spray (500 gpm) and containment venting (0.1 ft^2). The containment failure pressure (median value in Figure 6.27) is not reached for this case.

Table 6.1 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>	
	T/T > 95 F	(EPG Entry Condition)	Very Early Recovery Actions Monitor and Control T/T, T/L, D/P, and D/T (Note 1) <i>SP Cooling (Note 2)</i>
0:15			Declare Site Area Emergency Activate EOF
0:30	<i>O/P > 2 psig</i>	(TSC Operational) (EPG Entry Condition)	<i>CADS and SCIS</i> <i>DW Cooling & HW Spray</i>
1:00	T/T > 110 F	Potential Unacceptable SP Boundary Load	Enter EPG RPV Control G/L (Note 3)
1:30	T/T > 120 F	Same as Above	RPV Depressurization
5:00	T/T > HCTL	SP Boundary Load	Emergency RPV Depres.
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>	
11:00	Core Melt Starts RPV Repres. <i>O/T > 135 F</i> <i>O/P > 17 psig</i> <i>O/P Increase &</i> <i>VB imminent</i>	SP Boundary Load (EPG Entry Condition) SP Boundary Load Potential challenge at VB	Early Recovery Actions <i>Alternate E/P & P/S (Note 4)</i> <i>DW Cooling</i> <i>DW Spray</i> <i>DW Spray, Early HW Venting</i> <i>Reactor Cavity Flooding</i>
12:00	Vessel Breach	Load Associated with HPME (DCH, M&E Addition & SP Load) DSM	<i>RPV Dep. or Above Actions</i> <i>Before VB to be Effective</i> <i>DW Spray & Corium Flood.</i>

Table 6.1 (Continued)

Time Hr:Min	Plant Status	Challenge	Strategy or SAM Actions
12:00	D/P > 40 psig D/T > 281 F CCI	<u>Late Phase of the Accident</u> Containment Pressure Load Containment Temperature Load Pressure, Temperature Loads & Noncondensable Gas Generation	(Note 5) Late Recovery Actions RPV Depres. DW Spray RPV Depres.. Cont. Cooling, DW Spray Corium Flooding & Containment Flooding DW Spray Containment Venting
14:00	D/P > PCPL	Containment Pressure Load	
15:00	Containment Failure	<u>Release Phase of the Accident</u>	
11:00 - 12:00	In-Vessel Rel. & Release at VB	FP in Containment Atmosphere	RPV Depres.
12:00 -	Ex-Ves. Rel. (CCI) Revolatilization Late Iodine Rel. All Above		Corium Flooding Water to RPV & DW Cooling SP Cooling DW Spray and Containment Flooding
15:00 -	Cont. Failure or Venting	FP Release from Cont. to Outside	WW Venting Flooding Leak Area Containment Flooding
	FP & Pressure in RB	FP Release from RB to Environment	SGTS Fire Spray

Table 6.1 (Continued)

Note:

1. T/T - Torus (SP) Temperature, T/L - Torus Level, D/P - Drywell or Containment Pressure, D/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. Maintaining RPV depressurization using alternate electric power (E/P) and pneumatic supply (P/S) as recommended by CPI.
5. The availability of instruments and equipment after battery depletion is very uncertain. Unless special arrangements have been made, they are generally not available. However, recovery of E/P will make some of them available.

Table 6.2 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 Emerg. Declared General Emergency Declared Activate TSC & FOF
		<u>Very Early Phase of the Accident</u>	
	$I/T > 95 F$	(EPG Entry Condition)	Very Early Recovery Actions Monitor and Control T/T, T/L, D/P, and D/T (Note 1) <i>SP Cooling (Note 2)</i>
0:15	$I/T > 110 F$	Potential Unacceptable SP Boundary Load	Enter EPG RPV Control G/L (Note 3)
0:30	$I/T > 120 F$	(TSC Operational) Potential Unacceptable SP Boundary Load	<i>RPV Depressurization (Note 4)</i> <i>CADS and SCIS</i> <i>DW Cooling & WW Spray</i>
1:00	$D/P > 2 \text{ psig}$	(EPG Entry Condition)	
		<u>Early Phase of the Accident</u>	
2:00	Core Melt Starts RPV Repres. $I/T > \text{HCLL}$ $D/P > 17 \text{ psig}$ $D/P \text{ Increase \&}$ $VB \text{ imminent}$	SP Boundary Load Same as Above Same as Above Potential challenge at VB	Early Recovery Actions <i>Alternate E/P & P/S (Note 4)</i> Emergency RFV Depres. <i>DW Spray</i> <i>Early WW Venting</i> <i>Reactor Cavity Flooding</i> <i>DW Spray</i>
3:20	Vessel Breach	Load Associated with HPME (DCH, M&E Addition & SP Load) DSM	<i>RPV Dep. or Above Actions</i> <i>Before VB to be Effective</i> <i>DW Spray & Corium Flood.</i>

Table 6.2 (Continued)

Time Hr:Min	Plant Status	Challenge	Strategy or SAM Actions
3:20	D/T > 135 F D/P > 40 psi D/T > 281 F CCI	<u>Late Phase of the Accident</u> (EPG Entry Condition) Containment Pressure Load Containment Temperature Load Pressure, Temperature Loads & Noncondensable Gas Generation	(Note 5) Late Recovery Actions DW Cooling RPV Depres. DW Spray RPV Depres. Corium Flooding Containment Flooding DW Spray Containment Venting
5:20	D/P > PCPL	Containment Pressure Load	
6:00	Containment Failure	<u>Release Phase of the Accident</u>	
2:00 - 3:20	In-Vessel Relea & Release at VB	FP in Containment Atmosphere	RPV Depres.
3:20 -	Ex-Ves. Rel. (CCI) Revolatilization Late Iodine Rel. All Above		Corium Flooding Water to RPV & DW Cooling SP Cooling DW Spray and Containment Flooding
6:00 -	Cont. Failure or Venting	FP Release from Cont. to Outside	WW Venting Flooding Leak Area Containment Flooding
	FP & Pressure in RB	FP Release from RB to Environment	SGTS Fire Spray

Note: See Notes of Table 6.1.

Table 6.3 FP Release for Various Fast SBO Cases (See Figure 6.11)

a. Release Fraction by Groups

APB No.	Description	Release Fraction by Group								
		1 (Kr-Xe)	2 (I)	3 (Cs)	4 (Te)	5 (Sr)	6 (Ru)	7 (La)	8 (Ce)	9 (Ba)
45	Base Case	1.00	2.4E-02	2.8E-02	2.6E-01	4.6E-01	8.8E-07	3.5E-02	5.2E-02	3.1E-01
37	Early WW Vent	1.00	2.7E-03	4.8E-03	2.7E-02	1.8E-02	6.8E-07	1.3E-03	2.0E-03	1.3E-02
29	Late Spray	1.00	7.5E-03	1.6E-02	3.3E-02	9.0E-02	1.2E-07	4.9E-03	7.6E-03	4.8E-02
43	Fire Spray	1.00	4.8E-03	5.6E-03	5.2E-02	9.1E-02	1.8E-07	7.0E-03	1.0E-02	6.6E-02
30	Early DW Leak	1.00	5.0E-02	5.7E-02	2.5E-01	4.3E-01	7.4E-07	3.2E-02	4.7E-02	3.2E-01
39	Late Vent	1.00	3.9E-05	2.1E-06	5.9E-03	6.4E-04	1.4E-07	2.2E-04	2.8E-04	1.7E-03
50	Late CF	1.00	2.4E-05	1.8E-08	8.3E-03	7.6E-05	1.3E-08	3.4E-04	3.0E-04	1.5E-03
32	Late DW Leak	0.91	8.8E-06	5.7E-09	3.5E-03	2.5E-05	5.3E-09	1.3E-04	1.1E-04	5.7E-04

b. Equivalent Fraction of a Single Radionuclide

APB No.	Description	Equivalent Fraction of (See Table 4.11)			Equivalent Fraction of Noble Gas at 2 Hrs (Table 4.12)	
		Iodine for Early Effect	Cesium for Late Effect			
45	Base Case	1.723	1.558		129.2	
37	Early WW Vent	0.160	0.069		6.4	
29	Late Spray	0.341	0.263		20.9	
43	Fire Spray	0.409	0.320		26.4	
30	Early DW Leak	1.667	1.559		121.2	
39	Late Vent	0.091	0.009		1.7	
50	Late CF	0.094	0.008		1.9	
32	Late DW Leak	0.078	0.004		1.2	

Note: Reactor building is bypassed in APB 30 and APB 37.

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>	
	T/T > 95 F	(EPG Entry Condition)	Very Early Recovery Actions Monitor and Control T/T, T/L, D/P, and D/T (Note 1) SP Cooling
	T/T > 110 F	Potential Unacceptable SP Boundary Load	Enter EPG RPV Control G/L (Note 2)
	T/T > 120 F	Same As Above	RPV Depressurization
0:15	D/P > 2 psig	(EPG Entry Condition)	CADS and SGTS DW Cooling & WW Spray Emergency RPV Depres.
	T/T > HCTL	SP Boundary Load	
0:30	D/T > 135 F	(TSC Operational) (EPG Entry Condition)	DW Cooling
	D/P > 17 psig	SP Boundary Load	DW Spray
1:00	D/P > 40 psig	Containment Pressure Load	RPV Depres.
	D/T > 281 F	Containment Temperature Load	DW Spray RPV Depres.
1:15	D/P > PCPL	Containment Pressure Load	DW Spray Containment Venting
1:30	Containment Failure	Release of Containment Atmosphere	(See Release Phase)
		<u>Early Phase of the Accident</u>	
2:15	Core Melt Starts		Early Recovery Actions
3:50	Vessel Breach	DSM	DW Spray & Corium Flood.

Table 6.4 (Continued)

Time Hr:Min	Plant Status	Challenge	Strategy or SAM Actions
3:50	CCI	<u>Late Phase of the Accident</u> Noncondensible Gas Generation	Late Recovery Actions Corium Flooding Containment Flooding
1:30 - 2:15	Containment Failure	<u>Release Phase of the Accident</u> Release of Containment Atmosphere	Resource Management
2:15 - 3:50	In-Vessel Rel. & Release at VB	FP in Containment Atmosphere	RPV Depres.
3:50 -	Ex-Ves. Rel. (CCI) Revolatilization Late Iodine Rel. All Above		Corium Flooding Water to RPV & DW Cooling SP Cooling DW Spray and Containment Flooding
2:15 -	Cont. Failure or Venting	FP Release from Cont. to Outside	WW Venting Flooding Leak Area Containment Flooding
	FP & Pressure in RB	FP Release from RB to Environment	SGTS Fire Spray

Note:

1. T/T - Torus (SP) Temperature, T/L - Torus Level, D/P - Drywell or Containment Pressure, D/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>	
	T/T > 95 F	(EPG Entry Condition)	Very Early Recovery Actions Monitor and Control T/T, T/L, D/P, and D/T (Note 1). SP Cooling
	T/T > 110 F	Potential Unacceptable SP Boundary Load	Enter EPG RPV Control G/L (Note 2)
	T/T > 120 F	Same As Above	RPV Depressurization
0:15	D/P > 2 psig	(EPG Entry Condition)	CADS and SGTS DW Cooling & WW Spray Emergency RPV Depres.
	T/T > HCTL	SP Boundary Load	
	D/T > 135 F	(TSC Operational) (EPG Entry Condition)	DW Cooling
	D/P > 17 psig	SP Boundary Load	DW Spray
		<u>Early Phase of the Accident</u>	
1:00	Core Melt Starts		Early Recovery Actions
	D/P > 40 psig	Containment Pressure Load	RPV Depres.
	High Pressure VB imminent	Potential challenge at VB	Early WW Venting Reactor Cavity Flooding DW Spray
2:00	Vessel Breach	Load Associated with HPME (DCH, M&E Addition & SP Load) DSM	RPV Dep. or Above Actions Before VB to be Effective DW Spray & Corium Flood.

Table 6.5 (Continued)

Time Hr:Min	Plant Status	Challenge	Strategy or SAM Actions
2:00		<u>Late Phase of the Accident</u>	
	D/T > 281 F	Containment Temperature Load	Late Recovery Actions DW Spray RPV Depres.
	D/P > PCPL	Containment Pressure Load	DW Spray Containment Venting
	CCI	Pressure, Temperature Loads & Noncondensable Gas Generation	Corium Flooding Containment Flooding
2:00	Containment Failure		
1:30 -	In-Vessel Rel.	<u>Release Phase of the Accident</u> FP in Containment Atmosphere	RPV Depres.
2:00	& Release at VB		
2:00 -	Ex-Ves. Rel. (CCI) Revolatilization Late Iodine Rel. All Above		
2:00 -	Cont. Failure or Venting		
2:00 -	FP & Pressure in RB	FP Release from RB to Environment	Corium Flooding Water to RPV & DW Cooling SP Cooling DW Spray and Containment Flooding WW Venting Flooding Leak Area Containment Flooding SGTS Fire Spray

Note: See Notes of Table 6.4.

PEACH BOTTOM TBI

Figure 6.1 Containment pressure response for TB1 (Reference 16)

Figure 6.2 Containment atmosphere temperature response for TB1 (Reference 16)

PEACH BOTTOM TBUX2

Figure 6.3 Primary containment pressure response - Peach Bottom TBUX (Reference 39)

Figure 6.4 Primary containment temperature response - Peach Bottom TBUX (Reference 39)

Figure 6.5 Drywell Pressure for SBO Sequences

Figure 6.6 Drywell Temperature for SBO Sequences

Figure 6.7 Drywell Pressure for various Drywell Spray Flowrates (SBO with spray initiation at 205 minutes)

Figure 6.8 Drywell Temperature for various Drywell Spray Flowrates (SBO with spray initiation at 205 minutes)

Figure 6.9 Drywell Pressure for various Vent Areas
(SBO with vent initiation at 115 psia, no reclosure)

Figure 6.10 Drywell Temperature for various Vent Areas
(SBO with vent initiation at 115 psia, no reclosure)

Figure 6.11 Partial Release CRET for FP Release Comparison (For Table 6.3)

	CF Time	CF Mode	Wetwell Venting	Late DW Spray	Fire Spray	APB No. (Ref. 54)
						37
						29
		Rupture				43
					Base Case	45
						30
						39
						50
						32

PEACH BOTTOM TC1

Figure 6.12 Containment Pressure Response for TC1(Reference 16)

Figure 6.13 Containment Atmosphere Temperature Response For TC1 (Reference 16)

Figure 6.14 Drywell Pressure for various Vent Areas
ATWS with ECCS pumps running throughout accident

Figure 6.15 Drywell Temperature for various Vent Areas
ATWS with ECCS pumps running throughout accident

Figure 6.16 Non-condensable Partial Pressure in Drywell for various Vent Areas
ATWS with ECCS pumps running throughout accident

Figure 6.17 Non-condensable Partial Pressure in Wetwell for various Vent Areas
ATWS with ECCS pumps running throughout accident

PEACH BOTTOM TC2

Figure 6.18 Containment Pressure Response for TC2 (Reference 16)

PEACH BOTTOM TC2

Figure 6.19 Containment Atmosphere Temperature Response for TC2 (Reference 16)

Drywell Pressure for various Vent Areas

ATWS with ECCS pumps failing, HPCI 200 F, RCIC 25 psia

Figure 6.20 Drywell Pressure for various Vent Areas
ATWS with ECCS pumps failing, HPCI 200 F, RCIC 25 psia

Figure 6.21 Pressures in Containment Volumes - Sequence TW (Reference 15)

Figure 6.22 Gas Temperatures in Containment Volumes - Sequence TW (Reference 15)

Figure 6.23 Pressures in Containment Volumes - Sequence AE (Reference 15)

Figure 6.24 Temperatures in Containment Volumes - Sequence AE (Reference 15)

PEACH BOTTOM V

Figure 6.25 Containment Pressure Response for V Sequence (Reference 16)

Figure 6.26 Containment Atmosphere Temperature Response for V Sequence (Reference 16)

Figure 6.27 Containment Failure Pressure and Temperature

Figure 6.28 Effects of Venting on Drywell Pressure and Pressure Capability

Figure 6.29 Effects of Drywell Spray on Drywell Pressure and Pressure Capability

Figure 6.30 Containment Pressure and Pressure Capability for Case with both Drywell Spray and Containment Venting

Figure 6.31 Drywell Temperature for Case of both Drywell Spray and Containment Venting

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 I 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 existing 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 I 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 I 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 containment 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., drywell shell meltthrough, there is no direct instrument indication and the identification of the challenge must rely on indirect indications.

7.2 Guidelines for Containment and Release Management (CRM)

Although the existing EPGs extend beyond the design bases and include many actions appropriate for severe accident management,

there are areas that could benefit from additional coverage. Separate guidelines, more flexible than the existing EPGs (because of the greater uncertainties in containment performance and phenomena under severe accident conditions) may be desirable for containment and release management. The containment and release event trees (CRETs) discussed in this report provide a framework for accident progress projection and CRM decision making. The use in the CRETs of current plant status and offsite information together with updated and more accurate estimates of the probability for recovery, etc. can provide a more reliable prediction of the effects of various CRM strategies on accident progression and offsite consequences. Such an approach can be the basis for optimum (in a probabilistic sense) containment and release management.

7.3 CRM Strategies

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

The results of the strategy identification effort are summarized in the safety objective tree (SOT) in figure 4.15. The important strategies identified by this investigation are presented in Table 5.2. Although some of the strategies (e.g., wetwell 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 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 Peach Bottom plant. The strategies should be beneficial for other BWR plants with Mark I 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 under-

standing of containment performance, and the ability to predict accident progression accurately. For some strategies further investigation at this time may not be warranted until phenomena are better understood. 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 determine such optimum choice of strategies.

7.4 An Integrated Approach for CRM

CPETs 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 tree, can be used to quantify the effectiveness of individual strategies. Another application of the CPETs 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)

This report identifies and assesses accident management strategies which could be important for preventing containment failure and/or mitigating the release of fission products during a severe accident in a BWR plant with a Mark I type of containment. Based on information available from probabilistic risk assessments and other existing severe accident research, and using simplified containment and release event trees, the report identifies the challenges a Mark I containment could face during the course of a severe accident, the mechanisms behind these challenges, and the strategies that could be used to mitigate the challenges. A safety objective tree is developed which provides the connection between the safety objectives, the safety functions, the challenges, and the strategies. The strategies were assessed by applying them to certain severe accident sequence categories which have one or more of the following characteristics: have high probability of core damage or high consequences, lead to a number of challenges, and involve the failure of multiple systems.

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