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PREVENTION AND MITIGATION OF SEVERE ACCIDENTS IN A BWR-6 WITH A MARK III CONTAINMENT

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ABSTRACT

Preliminary guidelines and proposed criteria have been developed for the prevention and mitigation of severe accidents in a BWR-6 reactor with a Mark III containment. The preliminary guidelines were developed from insights derived from reviews of in-depth risk assessments performed specifically for the Grand Gulf plant and from other relevant studies. Accident sequences that dominate the core damage frequency and those accident sequences that are of potentially high consequence were identified. Vulnerabilities of the Mark III containment to severe accident containment loads were also identified. In addition, those features of a BWR-6 with a Mark III containment, which are important for preventing core damage and are available for mitigation of fission product release to the environment were also identified. Based on this information, preliminary guidelines with associated proposed criteria were developed.

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PREFACE

This draft report addresses the subject of formulation of guidelines and criteria for severe accidents in BWR-6 reactors with Mark III containments. It is an interim (preliminary) product of a technical assistance contract with NRC/NRR in support of their Implementation Plan for the Severe Accident Policy Statement (see SECY-86-76, February 28, 1986 for details of this plan). It is important to emphasize that while this effort required a broad range of in-depth expertise from Brookhaven National Laboratory (BNL) in the area of plant systems and operations, accident sequence analysis, severe accident phenomenology, and risk integration, there was considerable input from the NRR staff on program emphasis and technical direction. In particular, BNL was requested by the staff to formulate preliminary guidelines and proposed criteria that are deterministic (rather than probabilistic) in character.

The information contained in this draft is subject to revision upon receipt of information from two other programs. The IDCOR program, sponsored by the nuclear utility industry, is developing a methodology for individual plant examinations (IPE) which, subject to evaluation and modification by NRC, would be used in conjunction with guidelines and criteria developed by NRC. BNL has performed a preliminary review of the IDCOR IPE methodology and many of the criteria proposed herein reflect insights from the IDCOR methods. The SARRP program, sponsored by the NRC Office of Nuclear Regulatory Research, is re-baselining risk for several reference plants and this will be published in NUREG-1150. BNL has received preliminary results on portions of this work along with a caveat from the SARRP contractor that the results are subject to revision.

The reviewers of this draft report are encouraged to provide comments and suggestions on all aspects of this work.

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NOMENCLATURE

A	Large Loss of Coolant Accident (LOCA)
ADS	Automatic Depressurization System
ARC	Alternate Room Cooling
ARI	Alternate Rod Insertion
ASEP	Accident Sequence Evaluation Program
ATWS	Anticipated Transient(s) Without Scram
BCL	Battelle Columbus Laboratories
BF	Browns Ferry Nuclear Station
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
C	Failure of Reactor Protection System (RPS)
CDEP	Failure of Manual Depressurization
CDF	Core Damage Frequency
CHR	Containment Heat Removal
CRD	Control Rod Drive System
C_1	Mechanical Failure to Scram
C_2	Operator Failure to Actuate Standby Liquid Control System (SLC) or to Control Level with High Pressure System (HPS), or Failure of SLCS
DG	Diesel Generator
DGCM	Diesel Generators Common Mode Failure
DGREC	Failure to Recover Diesel Generators
DHR	Decay Heat Removal
E	Failure of Coolant Injection
ECCS	Emergency Core Cooling Systems
EPG	Emergency Procedure Guidelines
ESWS	Emergency Service Water Systems
FW	Feedwater System
GG	Grand Gulf Nuclear Station
HADS	Failure to Inhibit ADS
HEP	Human Error Probability
HPCI	High Pressure Coolant Injection System
HPIS	High Pressure Injection Systems

HPLC	Failure to Control RPV Water Level with HPCI during ATWS (either due to Operator Error and/or Hardware Failure or Malfunction)
HPSW	High Pressure Service Water System
I	Failure of Containment Heat Removal
IDCOR	Industry Degraded Core Rulemaking Program
INJ	Failure of Injection with Low Pressure Systems (LPS) after Containment Failure (CF)
IORV	Inadvertent Open Relief Valve
IREP	Interim Reliability Evaluation Program
J	Failure of the HPSW
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power (sometimes denoted by LOSP)
LPSI	Low Pressure Safety Injection Systems
LPLC	Failure to Control RPV Water Level at Low Pressure during ATWS (either due to Operator Error and/or Hardware Failure or Malfunction)
MSIV	Main Steam Isolation Valve
P	One or More Stuck Open Relief Valves (SORV)
PB	Peach Bottom Atomic Power Station
PCS	Power Conversion System
Q	Failure of Feedwater System
RCIC	Reactor Core Isolation Cooling System
RHR	Reactor Heat Removal System
RPV	Reactor Pressure Vessel
RSS	Reactor Safety Study
RSSMAP	Reactor Safety Study Methodology Application Program
S	Small LOCA
SARP	Severe Accident Research Program
SARRP	Severe Accident Risk Reduction Program
SLC	Failure of SLCS (due to Failure of Manual Initiation and/or due to Hardware Malfunction or Failure)
SLCS	Standby Liquid Control System
SNL	Sandia National Laboratories
SORV	Stuck Open Safety Relief Valve
SW	Service Water

(*)	Transient sequences are denoted by T followed by letters denoting the relevant failure, e.g., TC transients involving failure of RPS TQUV transients involving failure of FW, HPIS, and LPIS, etc.	
T	Transient	
TAF	Top of Active Fuel	
TB	Station Blackout Sequence (sometimes referred to as SBO)	
T_1	Loss of Offsite Power (LOOP) Initiator	
T_{23}	All Other Transient Initiators Except LOOP	
U	Failure of High Pressure Injection Function	
V	Failure of Low Pressure Injection Function	
VCS	Operator Failure to Control Level and Reactivity with Low Pressure Systems	
VENT	Containment (Wetwell) Venting Failure	
W	Failure of Containment Heat Removal (CHR)	
X	Failure of Reactor Pressure Vessel (RPV) Depressurization	

1. EXECUTIVE SUMMARY

This investigation was performed in support of the NRC/NRR Implementation Plan (SECY-86-76) for the Severe Accident Policy Statement. Based on an extensive review of severe accident investigations, the authors have proposed a set of preliminary guidelines and associated detailed criteria which can be used to assess the capability of individual BWR-6, Mark III plants to cope with severe accidents. Although much of the work is based on probabilistic risk assessments (PRAs), the preliminary guidelines and criteria are deterministic in nature and take into account detailed severe accident experiments and analyses performed by both NRC/RES and the nuclear industry.

1.1 Core Damage Profile

Appendix A provides a review of BWR-6 risk assessment studies with emphasis on the recent ASEP and IDCOR results. The Grand Gulf RSSMAP study and the GESSAR II PRA are also included. Transients rather than LOCAs dominated the core damage risk profile for the studies examined. There was no consistent pattern of relative ranking of transient sequences across all of the studies. However, in the later studies the same few functional accident sequences figured prominently in the core damage frequency profiles. It is also important to observe that for a given accident sequence, the major contributor to differences in quantitative results between the studies was due to subjective modeling assumptions rather than plant differences or data differences.

For the RSSMAP study of Grand Gulf, loss of containment heat removal sequences (e.g., TQW and TPQI) appeared as important contributors to the core damage frequency (about 50%). Although the more recent studies have reduced the core melt frequency due to these sequences based on operating procedures for alternate injection, detailed criteria have been developed to ensure that these sequences are not dominant for other BWR Mark III plants.

Both ASEP and IDCOR indicated that station blackout and ATWS are the dominant core damage sequences for Grand Gulf. Both studies calculate a total core damage frequency (CDF) only slightly higher than 10^{-6} per reactor year.

Thus, the preliminary guidelines for other BWR Mark III plants will attempt to ensure that the likelihood of these "dominant" sequences are kept at a correspondingly low level.

1.2 Consequence Analysis

The assessment of core meltdown phenomena and containment response in Appendix A indicated that the Mark III containment is vulnerable to severe accident containment loads. Unless mitigative actions are taken a Mark III containment has the potential to fail a short time (a few hours or less) after the reactor vessel fails. However, both IDCOR and ASEP/SARRP predict the containment failure (if it occurs) location to be the outer wall. Therefore, fission products released into the drywell will still pass through the pool. Thus, even with a containment failure, containment function (reduction of the source term) is preserved for almost all cases. Only direct bypass sequences (interfacing system LOCA) or drywell wall failure result in severe releases of fission products.

1.3 Proposed Guidelines

The assessment of core meltdown phenomena and containment response indicates that the Mark III containment provides a vigorous defense against fission product release even under severe accident loads. The ability of the Mark III suppression pool to trap aerosol fission products is an important mitigative feature since it leads to a direct reduction in offsite consequences by a factor of 10 or more. Thus, any pathways that might open, which would allow the fission products to bypass the pool are undesirable. The following are possible ways in which the suppression pool may be bypassed:

- failure of vacuum breakers between the drywell and wetwell

- failure of drywell penetrations due to high temperature

- structural failure of the drywell due to hydrogen explosions

- collapse of the drywell wall as a result of contact with molten core materials

- LOCA through a low pressure system outside containment due to failure of the pressure isolation valves

Because of the importance of the suppression pool as a mitigative feature, the vulnerability of a Mark III containment to any of the above bypass pathways should be examined. However, none of the studies reviewed in Appendix A has identified these failure mechanisms as being a significant threat for Mark III plants. Guideline 1 is proposed to ensure the low frequency of interfacing system LOCA events based on insights from PRAs for other plant types.

The most important contributors to CDF for Grand Gulf were found to be station blackout and ATWS sequences. However, other accident sequences that were not important contributors to CDF in Grand Gulf could become dominant for other Mark III plants due to the unavailability of certain mitigating features that were available in Grand Gulf. Therefore, the proposed guidelines are intended to cover a large spectrum of accident sequences. In the ATWS sequences, the containment will pressurize rapidly and may fail with resultant loss of coolant injection and eventual core melt into a failed containment. Containment venting has the potential to mitigate the containment failure; however, the rapid progress of ATWS events restricts the likelihood of successful venting. Thus, Guideline 2 is proposed to ensure the low frequency of occurrence of ATWS events.

Guideline 3 addresses accidents involving the loss of offsite power and onsite emergency power and assumes compliance with the proposed station blackout rule with respect to reducing their frequency of occurrence. Guideline 3 proposes the means to mitigate the consequences of proposed station blackout sequences by requiring the major mitigating features (high pressure injection and hydrogen control systems) to have diversity and independence from their current station emergency power supplies as well as alternate cooling capability for the high pressure injection systems.

Accident sequences involving loss of containment heat removal (e.g., TW) were found to be quite important in the earlier PRA studies addressed in Appendix A. In the case of WASH-1400, the TW sequences accounted for 53% of the calculated core damage frequency. In the Grand Gulf RSSMAP study, the TW sequences similarly accounted for about 50% of the calculated core damage frequency. The most recent Grand Gulf studies, IDCOR and ASEP/SARRP, show a two and three order of magnitude reduction, respectively, in TW sequences quantification. Therefore, Guideline 4 has been developed to generically address the mechanisms already effectively employed at Grand Gulf in reducing the TW sequences (and other related loss of containment heat removal sequences) from a dominant sequence for the BWR-6, Mark III designs to a point where both IDCOR and ASEP/SARRP indictate that it is an insignificant contributor to risk.

Finally, although the importance is difficult to quantify, one of the insights of most risk assessment studies is the importance of support system interdependencies. For example, a preliminary draft of the ASEP Peach Bottom study indicated that loss of all service water was a dominant contributor to core damage. The recent revision to the sequence studies have reduced it to one percent of the overall core damage. In order to ensure that support system vulnerabilities do not cause unacceptably high CDFs for BWR-6 Mark III plants, the authors have proposed Guideline 5 to help assess any weaknesses of the support systems.

2. INTRODUCTION

2.1 Background

During the next two years, the NRC plans to formulate an approach for a systematic safety examination of existing plants to determine whether particular accident vulnerabilities are present and what cost-effective changes are desirable to ensure that there is no undue risk to public health and safety.

At the same time, the Industry Degraded Core Rulemaking (IDCOR) Program has selected four reference plants for detailed analysis, namely:

Peach Bottom (a BWR with a Mark I containment)

Grand Gulf (a BWR with a Mark III containment)

Zion (a PWR with a large dry containment)

Sequoyah (a PWR with an ice condenser containment).

The IDCOR analyses performed for the above reference plants have been documented together with the methodology used for the analyses and the technical basis supporting the methodology. In addition, IDCOR is presently working on a simplified approach to be used for the safety examination of other plants similar to one of the reference plants. The simplified approach together with a few sample applications has been submitted for NRC review in draft form.

Parallel with the IDCOR work, NRC/RES under the Severe Accident Research Program (SARP) is performing risk assessments, audit calculations, sensitivity studies, and uncertainty analyses for six plants. The six plants to be considered include the above four IDCOR reference plants, and, in addition:

Surry (a PWR with a subatmospheric containment)

LaSalle (a BWR with a Mark II containment).

NRC/NRR is responsible for comparing both the IDCOR and SARP analyses performed for the reference plants, resolving differences, passing judgment on the adequacy of these plants with respect to public safety and then using the experience gained from these reviews for the development of specific guidelines and criteria for the prevention and mitigation of severe accidents for each plant type. In turn, these guidelines will be used in the systematic safety examination of individual plants. In addition, a review of the simplified approach for individual plant reviews being developed by IDCOR will also be part of the effort. BNL is under contract to NRC/NRR to assist in this effort.

Part of this work is nearing completion. The first plant reviewed was Peach Bottom,[1] which is a BWR-4 with a Mark I containment. The IDCOR Peach Bottom analysis was documented in March 1985 (IDCOR Technical Report 23.1PB) and supplemented by additional sensitivity studies in July 1985. However, information available from the Accident Sequence Evaluation Program (ASEP) at SNL at the time of the review was preliminary and subject to change. In spite of the preliminary nature of the ASEP analysis, the experience gained from the review of both studies was sufficient to generate preliminary guidelines and proposed detailed criteria.[1] This draft report therefore builds on the experience gained during our Peach Bottom work and on the comments received from reviewers of Reference 1. Specifically, this draft report deals with potential severe accidents in a BWR-6 with a Mark III containment. Both IDCOR and SARP used Grand Gulf as the reference plant for this class of reactors so this draft report is based largely on analyses of severe accidents at Grand Gulf. The IDCOR Grand Gulf analysis[2] was documented in March 1985 whereas the SARP analysis is again preliminary and subject to change.

2.2 Objectives

There are three basic objectives or goals for this severe accident program which will apply equally to all plant types:

Goal 1: Mitigation of fission product releases
Goal 2: Prevention of high consequence sequences
Goal 3: Prevention of high core damage frequency.

The aim is, therefore, to develop detailed plant type specific guidelines and proposed criteria to be used to achieve these goals during the examination of individual plants. For example, Goal 1 implies that there shall be effective means of mitigating the fission product releases for the broad classes of accident sequences which dominate the core damage frequency. Therefore, these dominant accident sequences have to be determined and those plant features and operator actions that are available to mitigate fission product release have to be identified. Only then can detailed guidelines and criteria be developed to ensure mitigation of these dominant accident sequences.

There may be accident sequences for which fission product release mitigation systems are impaired (e.g., containment bypass sequences). Thus, for these sequences Goal 1 may be difficult to achieve. Therefore, all reasonable steps should be taken to reduce the frequency of these potentially high consequence sequences (namely Goal 2). Again, the accident sequences have to be identified and plant vulnerabilities and/or operator actions that lead to core damage for these sequences also have to be identified. Detailed guidelines and criteria can then be developed which will aid in assessing an individual plant's capability to prevent these sequences from occurring.

Finally, it is necessary to ensure that the overall core damage frequency is low (namely Goal 3). Again, the dominant accident sequences have to be found so that detailed guidelines and criteria can be developed to reduce the frequency of these sequences, if necessary.

2.3 Organization of the Report

Appendix A contains a review of the IDCOR and ASEP/SARRP analyses for a BWR-6 with a Mark III containment along with other pertinent studies. The insights gained from these studies lead to the identification of the strengths and potential vulnerabilities of a BWR-6 with a Mark III containment. The three basic goals of the program are then related to the relevant design features and operating characteristics of a BWR-6 with a Mark III containment in Section 3. The preliminary guidelines necessary to achieve the three goals are therefore initially developed in Section 3. Finally, in Section 4 the preliminary guidelines are restated and detailed criteria are developed for each guideline.

2.4 References for Section 2

1. W. T. Pratt et al., "Prevention and Mitigation of Severe Accidents in a BWR-4 with a Mark I Containment," Draft BNL Technical Report A-3825R, August 8, 1986.

2. IDCOR Technical Report 23.1GG, March 1985.

3. DEFINITION OF GOALS AND RELEVANT BWR MARK III FEATURES

In Section 2 of this report, the concept of three basic objectives or goals for this severe accident program was introduced. The concept applies equally to all plant types. In this section, the three goals are related to the relevant design features and operating characteristics of a BWR-6 with a Mark III containment for the accident sequences and containment failure modes found to be important in Appendix A. This includes consideration of both favorable and unfavorable severe accident attributes. Screening criteria have been used to identify those sequences which need to be addressed by severe accident guidelines for each goal. Specifically:

For Goal 1 (Mitigation of fission product releases), all sequences have been examined which represent at least 5% of the core melt frequency or are estimated to occur more often than 10^{-6} per reactor-year.

For Goal 2 (Prevention of high consequence sequences) all sequences have been examined which result in pool bypass and are estimated to occur more often than 10^{-7} per reactor-year.

For Goal 3 (Prevention of high core damage frequency) all sequences have been examined which "have the potential to occur" more frequently than 10^{-6} per reactor year. Note that this screening criterion has been used to identify potential vulnerabilities from risk assessment insights which do not necessarily apply to Grand Gulf itself, but may apply to other Mark III plants.

This section provides the link between the goals (developed in Section 2) and the preliminary guidelines (developed in Section 4) that will be used to assess the capability of specific plants to meet these goals. This section is organized into three subsections, which correspond to the three goals.

3.1 Mitigation of Fission Product Releases

This goal requires that there shall be effective means of mitigating the fission product releases for the broad classes of accident sequences which may lead to core damage in a BWR-6 with a Mark III containment. In Appendix A, the most important contributors to the core damage frequency were found to be station blackout sequences and ATWS. Other transients and LOCAs may also contribute to the core damage frequency. Two specific accident sequences for which mitigation by the Mark III containment is ineffective were also identified in Appendix A. These specific sequences are discussed in Section 3.2, which attempts to determine how the frequency of these unmitigated sequences can be reduced. In this section, concentration on the broad classes of accident sequences for which plant features provide significant mitigation of the fission product releases will be made. In the following sections both the favorable and unfavorable severe accident attributes of the Mark III containment will be identified.

3.1.1 Plant Vulnerabilities

As noted in Appendix A, the Mark III containment is a pressure suppression design. The suppression pool is available to condense steam released from the primary system during an accident. However, the Mark III containment may be vulnerable to pressure/temperature buildup due to the noncondensible gases generated during a core meltdown accident. There are differences between the IDCOR and BCL (NUREG-1150) analyses as to how long it will take to pressurize a Mark III containment to its ultimate capacity after the core debris has failed the reactor vessel (and is interacting with concrete) but both studies concluded that containment failure will eventually occur. Therefore, unless mitigative actions are taken, a Mark III containment will fail eventually due to overpressure or overtemperature. If drywell leakage occurs, a fraction of the fission products in the drywell atmosphere could pass to the outer containment (and ultimately to the environment) without the benefit of suppression pool scrubbing. Note that suppression pool scrubbing is an important mitigative feature of a Mark III containment (refer to Section 3.1.2).

An inspection of the Mark III containment configuration in Appendix A (Figure A.2) will show that the pedestal below the reactor pressure vessel would tend to confine the core debris after a core meltdown accident. In the absence of a water supply (no water from a LOCA, upper pool dump or restoration of reactor coolant injection), extensive core/concrete interactions would be expected to occur. There are differences between the IDCOR and SARP analyses as to how hot the core debris will remain during these interactions and as to how many of the less volatile fission products will be released. IDCOR assumes that water from the lower plenum and CRD pumps will be available to quench the core debris and keep it cool as long as CRD flow is available. However, at this time, the authors do not believe that the possibility of the core debris remaining hot and releasing significant quantities of fission products has been ruled out particularly under station blackout conditions.

After the region directly underneath the reactor vessel (pedestal region) fills with core debris there would still be sufficient core materials in a full core meltdown to fill the cavity. If the core debris remains molten it could erode the pedestal support and cause the vessel to be displaced resulting in failure of the drywell wall. This is a mechanism for early loss of drywell integrity and is thus another Mark III containment vulnerability relative to some other containment designs in which the geometry would tend to disperse the core debris or quench it in the pool.

Both IDCOR[1] and BCL[2] predict that a substantial quantity of hydrogen will be produced during core degradation and core/concrete interaction. The hydrogen provides both a temperature and a pressure threat to containment. If the hydrogen burns, the high temperatures and pressures provide a threat to drywell integrity which may lead to pool bypass (as modeled in the SARP analysis). The Grand Gulf containment is equipped with hydrogen igniters which are intended to ensure that the hydrogen does not accumulate to explosive concentrations. However, the igniters depend on AC power. Therefore, they would not be available during blackout sequences. Even if the igniters perform their intended function, the resulting high temperatures may contribute to drywell penetration failure.

In the following section, suppression pool scrubbing is noted as an effective mitigative feature for the Mark III containment provided all of the fission products pass through the pool. It is, therefore, important to ensure that paths do not open which would allow the fission products to bypass the suppression pool. There are vacuum breakers between the wetwell and drywell that would result in a path which bypasses the suppression pool if they fail open. In addition, the various drywell penetration seals could be degraded at high temperatures and pressures. Failure of these seals would also open up paths which could bypass the suppression pool.

Although the Mark III containment appears to be vulnerable to high temperatures' high pressures and hydrogen combustion, they have several very important mitigative features, which are described in the following section.

3.1.2 Mitigating Features

The suppression pool in a Mark III containment is a very effective mechanism for trapping any fission product aerosols that might pass through it. Thus, to a large extent, the suppression pool has the potential to compensate for the vulnerabilities identified above (in Section 3.1.1). For example, overpressure failure of the containment can be prevented by venting. Containment integrity is lost but the containment function (retention of the fission products in the pool) is maintained.

High wetwell temperatures and possible drywell penetration seal degradation can be prevented by containment spray. Containment spray will also contribute to decontamination of the wetwell even for sequences with substantial pool bypass.

The Mark III containment has hydrogen igniters which prevent hydrogen accumulation. This is a very significant mitigative feature, which is important to maintain during a severe accident. However, the igniters, as currently powered, are not available during a station blackout.

The above discussion has identified several plant features of the BWR-6 plant with a Mark III containment that have the potential to help achieve Goal 1, namely, the mitigation of fission product releases. Moreover, both IDCOR and SNL indicate (see Appendix A) that significant bypass (beyond design leakage) of the pool is very unlikely. With a low probability (<10%) of early pool bypass, additional mitigative guidelines and criteria do not appear to be justifiable and we have therefore not developed any guidelines in Section 4 to meet Goal 1. A relatively low likelihood of pool bypass is also indicated in the GESSAR PRA[3] and the Safety Evaluation[4] of GESSAR.

3.2 Prevention of High Consequence Sequences

The plant features identified in Section 3.1 (the suppression pool, containment sprays and hydrogen igniters), have been found to effectively mitigate fission product release for the broad classes of accident sequences that were found to dominate the core damage frequency. However, accident sequences were found in Appendix A for which the BWR-6, Mark III plant may not be effective in mitigating fission product release.

The interfacing systems LOCA would open up a path from the primary system, bypassing the primary containment and suppression pool completely. The only plant feature pertinent to mitigating this sequence is the auxiliary building, which is not sufficient on its own to ensure low fission product release to the environment. The frequency of these potentially high consequence accident sequences must, therefore, be maintained at acceptably low levels (Goal 2). Neither IDCOR nor SNL have identified the interfacing systems LOCA as a significant contributor to core melt frequency. However, since the consequences of an interfacing system LOCA are potentially high and it is the subject of ongoing research (Generic Issue 105), a preliminary guideline has been developed pending resolution of the issue. This preliminary guideline and associated criteria related to Goal 2 dealing with prevention of high consequence sequences are developed in Section 4.

3.3 Prevention of High Core Damage Frequency

In Appendix A only a few accident sequences were found which figure prominently in the core damage profiles of all of the PRAs reviewed. This led to the conclusion that if the frequency of this relatively small subset of accident sequences can be controlled then the overall core damage frequency should also be controlled. In the following sections, these "dominant" core damage sequences are identified and discussed.

3.3.1 Station Blackout

The most important contributors to the core damage frequency were found to be station blackout and anticipated transients without scram (ATWS) sequences. Therefore, severe accident guidelines with specific detailed criteria have been developed in Section 4 related to these accident sequences. Station blackout is currently the subject of an unresolved safety issue (namely A-44). Thus, development of guidelines and criteria for station blackout must be considered preliminary pending resolution of A-44.

Station blackout refers to a loss of the offsite power system with concurrent failure of the two emergency AC power divisions. Reduction of station blackout sequences is addressed by the proposed NRC blackout rule. The findings of the present study indicate that there are measures that can be applied to the mitigating systems which could reduce the core damage frequency significantly. This is addressed in detail in Appendix A and is summarized in Table A.3.

Detailed criteria have been developed for the station blackout guideline that address the lowering of the core damage frequency by improving the response and long term survivability of the blackout mitigating function. For the BWR-6 design, the two systems designed to operate in the presence of a station blackout are the high pressure core spray (HPCS) and the reactor core isolation cooling (RCIC) systems. By removing the long term blackout sequence related to dependent failure modes of either system, the blackout core damage frequency can be significantly reduced.

3.3.2 Anticipated Transients Without Scram (ATWS)

ATWS has been identified as a potentially significant contributor to the core damage frequency in Appendix A. Therefore, a severe accident guideline has been developed in Section 4 related to these sequences. An ATWS rule has been recently issued and compliance with this rule was assumed in the formulation of the detailed criteria for this guideline.

The guideline for ATWS and the detailed accompanying criteria do not address specific hardware/systems modifictions as was proposed for the station blackout guideline. This is based upon the observations in Appendix A that a fairly large number of improvements to hardware/systems have already been developed and implemented in the BWR-6 design. Plants that have or plan to incorporate these design features will have acceptably reduced the ATWS core damage frequency without further hardware/systems modifications. It therefore must be stressed that the ATWS guideline and associated criteria in Section 4 assume that incorporation of the design features noted in Appendix A have or will be incorporated into the design (Alternative Rod Insertion (ARI) and high flow "equivalent" SLC).

3.3.3 Loss of Containment Heat Removal

Accidents involving loss of containment heat removal (CHR) were found to be important in the Grand Gulf RSSMAP report.[5] These accidents were found not to be important in the IDCOR and ASEP analyses for Grand Gulf because of credit given in these studies for wetwell venting (or other containment leakage) and alternative injection capability. Therefore, based upon engineering judgement, it has been deemed prudent to establish a guideline on this subject with attendant specific criteria. The underlying purpose of this guideline is to ensure that other Mark III plants will have the features/capabilities that validate the assumptions and credit given in the IDCOR and SARP analyses. Preliminary guidelines and detailed proposed criteria to ensure that loss of CHR sequences do not lead to core damage are developed in Section 4.

3.3.4 Support-System Interdependencies

Most PRAs have stressed the importance of unrecognized interdependencies having the potential to compromise the performnace of many critical safety systems. In many cases risk assessment studies have identified such vulnerabilities very early in the study and fixes have been made which substantially reduced risk. Although no such dependency caused vulnerability has been identified for Grand Gulf, "engineering judgement" indicates that such interdependencies should be identified for other Mark III plants.

3.4 References for Section 3

1. Grand Gulf Nuclear Station, "IDCOR Task 23.1 Integrated Containment Analysis," October 1984.

2. R. S. Denning et al., "Radionulcide Release Calculations for Selected Severe Accident Scenarios: BWR, Mark III Design," Battelle Columbus Laboratories, NUREG/CR-4624, Vol. 4, July 1986.

3. GESSAR (General Electric Standard Safety Analysis Report) II BWR/6 Nuclear Island, Probabilistic Risk Assessment.

4. GESSAR II SER, USNRC, NUREG-0979, April 1983.

5. S. W. Hatch et al., "Reactor Safety Study Methodology Applications Program: Grand Gulf #1 BWR Power Plant," NUREG/CR-1659/4 of 4, October 1981.

4. PRELIMINARY GUIDELINES AND CRITERIA

In Section 3 those accident sequences that dominate the core damage frequency were identified as were those that are potentially of high consequence. Vulnerabilities of the Mark III containment to severe accident containment loads were discussed and those features of a BWR-6 with Mark III containment, which are important for preventing core damage and available for mitigation of fission product release to the environment were identified.

Based on the "insights" from previous PRA studies, the following sections provide guidelines defining "deterministic, plant-specific guidance on the design features and operating characteristics which are to be examined by the utilities,"[1] and criteria defining "deterministic standards for judging the acceptability of plant features."[1] From SECY-86-76[2] further guidance is provided in defining preliminary guidelines and proposed criteria. These guidelines "will specify the plant features and operator actions which are considered important to ensuring acceptable risk for the reference plant."[2] Further acceptance criteria (for the various preliminary guidelines) "will specify the attributes necessary to ensure acceptable performance."[2]

Based on this work, five preliminary guidelines were developed which reflect the importance of these features to plant risk. The five preliminary guidelines are summarized in Table 4.0.

No preliminary guidelines were identified as being justified to be developed to ensure the capability to mitigate fission product releases (Goal 1) since current research indicates that the Mark III containment will provide sufficient mitigation.

One preliminary guideline was developed for the prevention of high consequence sequences (Goal 2) with reference to (1) minimization of interfacing systems LOCA frequency.

Also, four preliminary guidelines were developed to prevent a high overall core damage frequency (Goal 3) with reference to (2) mitigation of anticipated transient without scram (ATWS) sequences, (3) mitigation of station blackout sequences, (4) mitigation of loss of containment heat removal (CHR) sequences, and (5) analysis of support system interdependencies.

The remainder of this report is organized into three sections corresponding to the three basic goals. In each section, the corresponding preliminary guidelines are discussed from which detailed proposed criteria are developed in order to address the standards by which each plant should be measured to meet the severe accident guidelines. The criteria address the general issues of (a) operability and survivability of equipment and systems (i.e., the ability of the equipment to function under the environmental conditions and fluid dynamic loads associated with severe accident sequences), (b) capability and capacity of equipment, (c) reliability and accessibility of equipment, (d) availability of support systems, (e) identification of necessary components and operator actions, and (f) parameters for initiation of mitigating systems and operator actions.

4.1 Mitigation of Fission Product Releases

The review of containment performance for the dominant core melt sequences indicated that no preliminary guidelines were required to ensure the capability to mitigate fission product releases.

4.2 Prevention of High Consequence Sequences

Accident sequences were found in Appendix A for which the BWR-6, Mark III containment has limited means of mitigating fission product releases, namely an interfacing systems LOCA. In this section a preliminary guideline and proposed criteria for the prevention of these potentially high consequence sequences are developed. However, both IDCOR and SARP estimate these sequences to be small contributors to core melt for Grand Gulf. This guideline has been provided to ensure that other Mark III plants keep these high consequence sequences at a low level.

4.2.1 Minimization of Interfacing Systems LOCA Frequency (Preliminary Guideline 1)

As mentioned in Appendix A, BNL is presently performing a study to provide technical support to the NRC, for the meaningful resolution of the generic issue (GI-105) related to interfacing systems LOCA. Therefore, the criteria developed for this guideline in Table 4.1 should be considered as very preliminary.

To apply standards for the minimization of interfacing systems LOCA frequency guideline, the performance of equipment, systems, and operators should be assessed against specific performance criteria to ensure successful prevention of interfacing systems LOCA. The criteria relate to the equipment and operator performance as follows:

- low pressure systems interfacing with high pressure systems, and
- isolation valves and relief valves maintenance and surveillance.

4.3 Prevention of High Core Damage Frequency

The major contributors to the core damage frequency (CDF) were presented in Section 3.3. The IDCOR and ASEP/SARP analyses imply that the station blackout (SB) and ATWS sequences are the dominant contributors to the CDF. The results of other PRAs and PRA reviews indicate that in addition to those two types of sequences, other sequences, namely, loss of CHR sequences (TW, SI, TQUV, and TPQI sequences) and sequences with failure to depressurize the RPV for injection with low pressure systems (TQUX sequences), can also be major contributors to the core damage frequency. The difference between these results appears to arise from the assumption regarding whether loss of normal CHR leads to core damage.

4.3.1 Mitigation of Anticipated Transients Without Scram (ATWS) Sequences (Preliminary Guideline 2)

Five key actions are required by the operating crew to prevent core damage and/or containment failure during the worst ATWS sequence, namely with MSIV closure. These are:

- Initiate Standby Liquid Control (SLC) system immediately when suppression pool reaches initiation temperature. (Note that some Mark III plants have automatic SLC system initiation which do not require manual initiation unless the automatic initiation fails to operate.)

- Inhibit Automatic Depressurization System (ADS) after SLC initiation attempts.

- Maintain reactor pressure vessel (RPV) water level while at high pressure before depressurization.

- Manual depressurization using Safety Relief Valves (SRVs) when suppression pool temperature reaches the heat capacity limit curve.

- Maintain RPV level while at low pressure after depressurization.

The important attributes of this sequence with respect to operator actions were found[3] to be the likelihood of misleading instrumentation, the need to inhibit automatic initiated safety systems, the use of required mitigating actions which conflict with operator response to other accident conditions, and the need for coordinated actions and communication among control room crew members under highly stressful conditions.

To apply standards for the mitigation of ATWS sequences guideline, the performance of equipment, systems, and operators should be assessed against specific performance criteria to ensure successful accomplishment of this guideline. The criteria relate to the equipment, systems, and operator performance as follows:

- operator familiarization, aids and understanding of potentially conflicting signals.

Detailed criteria developed for this guideline are given in Table 4.2 and based upon the assumption that each of the plants are (or will be) in compliance with the NRC rule on "reduction of risk from ATWS for light-water-cooled nuclear power plants."[4]

4.3.2 Mitigation of High Station Blackout Sequences (Preliminary Guideline 3)

In most PRAs for LWRs, station blackout sequences have been prominent contributors to the CDF. The NRC is proposing to amend its regulations "to provide further assurance that a station blackout (loss of both offsite power and onsite emergency ac power systems) will not adversely effect the public health and safety."[5] The frequency of loss of offsite power, the reliability of the emergency ac system, and the ability of the plant to cope with a station blackout should be evaluated according to the method used for the NRC proposed rule on station blackout. Therefore, the criteria developed for this guideline in Table 4.3 should be considered preliminary pending final resolution of unresolved safety issue A-44.

The performance of equipment, systems and operators should be assessed against specific performance criteria to ensure successful accomplishment of this guideline. The criteria relate to the equipment, systems, and operator performance as follows:

- equipment needs with respect to cooling,
- equipment needs for dc power, and
- operator understanding of the above equipment needs and their limitations.

Due to the different configurations and different emergency ac and dc systems, the method used for the NRC proposed rule on station blackout may not necessarily address all plant-specific vulnerabilities of the emergency ac and dc power systems. Therefore, additional criteria may be necessary to address plant-specific vulnerabilities of the emergency ac and dc power systems, which may not be addressed in the method outlined in the NRC proposed rule. These vulnerabilities should include, but not be limited to, dependences on common support systems, lack of physical separation, common maintenance, and other plant-specific common causes of ac and dc unavailability.

4.3.3 Mitigation of Loss of Containment Heat Removal (CHR) Sequences (Preliminary Guideline 4)

In the Grand Gulf PSSMAP[6] report, sequences with successful coolant injection but with subsequent loss of containment heat removal (e.g., SI and TPQI sequences in Table A.1) were important contributors to the core damage frequency. In that study it was assumed that containment failure caused loss of injection. As discussed in Appendix A, in the PRAs where those sequences are not important, the main factor for the low contribution to core damage frequency is due to credit given for containment venting and alternative sources of injection. It would therefore appear that alternative injection sources should be available in addition to the wetwell venting to provide adequate CHR during accident sequences with successful coolant injection but with subsequent loss of CHR.

The performance of equipment, systems, and operators should be assessed against specific performance criteria to ensure successful prevention of core damage for loss of CHR sequences. The criteria relate to the equipment, systems, and operator performance as follows:

- source of cooling water,
- means of supplying the water,
- instrumentation and controls to monitor and direct the water, and
- operator aids, familiarization and expertise to initiate, control, and terminate the water.

Detailed criteria developed for this guideline are given in Table 4.4.

4.3.4 Analysis of Support-System Interdependencies (Preliminary Guideline 5)

One of the primary benefits of performing a rigorous PRA is that the system interdependencies are modeled and are reflected in the results. However, not all PRA studies have performed rigorous interdependence analyses and therefore may not have ferreted out all of the possible subtle interdependencies. This may have profound effects upon their results. An interdependency is defined as the failure of one system leading directly or indirectly to the failure of another system. A rigorous application of basic PRA methodology with respect to interdependencies yielded significant findings on a previously heavily studied plant.[7] These interdependency evaluation steps are outlined in Table 4.5.

It is not sufficient to make a single overall interdependency table of the front-line and support systems for a given plant and simply compare that to the reference plant. No two plants will have the same set of system interdependencies. Support systems vary widely from plant to plant even though the plants may be of a similar class and have the same set of front-line systems. It is recognized that following the steps outlined in Table 4.5, in a rigorous fashion, is a major undertaking. This fact, however, does not diminish its importance.

Based upon the dominance of the station blackout sequence to the BWR designs, it is recommended that a specific interdependency table be constructed for this sequence with all interdependencies conditioned upon the existence of a station blackout for various lengths of time. This table should also explicitly identify all of the expected failure mechanisms (e.g., identify whether battery failure is due to loss of room cooling or depletion).

4.3.5 Wetwell Venting Capability (Preliminary Guideline 6)

For sequences that threaten the containment by overpressure, wetwell venting has the potential to preserve the containment function by relieving non-condensible gases and/or saturated steam. Both IDCOR and SARRP results indicate that venting is not important to the release fraction. However, preserving the structural integrity of the containment is important to ensuring ECC injection system availability. For accident sequences resulting in loss of CHR prior to core damage, venting is a way of helping to prevent core damage. In addition, the possibility of failing the containment in the wetwell and allowing large fission product releases has not been precluded for all Mark III plants. Thus, it is strongly recommended that emergency procedures for wetwell venting be implemented.

For the small subset of ATWS sequences with uncontrolled low pressure injection, the resultant high power level appears to preclude venting and the containment spray would be isolated. The criteria necessary to control the ATWS core damage frequency are discussed in Section 4.3.1.

For the two dominant sequences of station blackout and ATWS, venting procedures will be difficult to perform. For station blackout sequences, power dependencies may preclude actuation of venting from the control room, and high radiation levels may hamper local manual actuation. For ATWS sequences, the large venting capacity requirements, short time frame for operator action and possible problems with normal isolation systems make successful venting under such conditions operationally difficult.

To apply standards for the preliminary guideline of wetwell venting, the appropriate expected equipment, system, and human performance must be assessed by appropriate criteria to ensure successful accomplishment of wetwell venting as required during severe accident conditions. The criteria relate to the venting equipment, systems, and human performance as follows:

- means to vent the wetwell,
- instrumentation and controls to monitor and direct venting
- operator familiarization and expertise to initiate, control, and terminate the venting.

Detailed criteria developed for this guideline are given in Table 4.6.

4.4 References for Section 4

1. R. Barrett, "Status of the Severe Accident Program for Operating Reactors," NRR Staff Presentation to the ACRS Subcommittee Class 9 Accidents, February 24, 1986.

2. SECY-86-76," Implementation Plan for the Severe Accident Policy Statement and the Regulatory Use of New Source-Term Information," NRC/EDO, February 28, 1986.

3. W. Luckas et al., "A Human Reliability Analysis for the ATWS Accident Sequence With MSIV Closure at the Peach Bottom Atomic Power Station," Technical Report A-3272 4/86, Brookhaven National Laboratory, April 1986.

4. ATWS Final Rule--Federal Register/Vol. 49, No. 124/Tuesday, June 28, 1984.

5. NRC Station Blackout Proposed Rule.

6. S. W. Hatch et al., "Reactor Safety Study Methodology Applications Program: Grand Gulf #1 BWR Power Plant," NUREG/CR-1659/4 of 4, Sandia National Laboratories, October 1981.

7. R. Youngblood et al., "Fault Tree Application to the Study of Systems Interactions at Indian Point 3," NUREG/CR-4207, January 1986.

8. M. T. Drouin et al., "Analysis of Core Damage Frequency from Internal Events: Grand Gulf Unit 1," NUREG/CR-4550, Vol. 6, Draft, Sandia National Laboratories, July 1986.

Table 4.0 Prelimnary Guidelines for the Prevention and Mitigation of Severe Accidents in a BWR-6 with a Mark III Containment

Preliminary Guideline	Description
<u>For Prevention of High Consequence Sequences:</u>	
1	Minimization of Interfacing Systems LOCA Frequency
<u>For Prevention of High Core Damage Frequency:</u>	
2	Mitigation of Anticipated Transient Without Scram
3	Mitigation of Station Blackout Sequences
4	Mitigation of Loss of Containment Heat Removal Sequences
5	Analysis of Support System Interdependencies
6	Maintenance of Containment Integrity

Table 4.1 Proposed Criteria for BWR Mark III Containment
Preliminary Guideline 1: Minimization of
Interfacing Systems LOCA Frequency

Concern: Although the interfacing system LOCA sequences have not shown themselves to be leading contributors to core damage frequency, they represent potentially high release sequences and they appear to contribute significantly to the overall risk for the plant under review.

Function: Maintain Primary System Integrity

Guideline 1. Minimization of Interfacing Systems LOCA Frequency

Basis: 1. Implementation of the following criteria will ensure the frequency of an interfacing system LOCA will remain acceptably low.

Criteria:

Note: Resolution of Generic Issue (GI-105), "Interfacing Systems LOCA at BWRs" may impact this guideline. Therefore, the critieria below should evaluate and factor in the proposed recommendations from the generic issue.

1.1. All low pressure lines that potentially could be overpressurized should be identified and should be provided with alarms to alert the operator of an overpressure event.

1.2. Operator training procedures should include specific instructions on what actions can be taken to isolate the low pressure systems identified in 1.1 above.

1.3. The pressure isolation valves designated to provide isolation and prevent overpressurization of low pressure systems should periodically undergo local leak rate testing (LLRT).

1.4. The relief valves designated to mitigate overpressurization should be demonstrated to be capable of relieving full primary system pressure at the corresponding maximum expected flow rates for each line.

1.5. Maintenance and surveillance procedures and related training should be consistent with manufacturer's recommendations and specify actions to be taken to ensure that the designated pressure isolation valves and relief valves are capable of performing as required.

1.6. After each reactor shutdown and cooldown, testing of the pressure isolation valves should be performed. Testing of these valves should not be performed under reactor operating conditions.

Table 4.2 Proposed Criteria for BWR Mark III Containment – Preliminary Guideline 2: Mitigation of Anticipated Transients Without Scram (ATWS) Sequences

Concern: ATWS sequences have been shown to be one of the leading classes of severe accident sequences in terms of core damage frequency for most BWRs. Although Grand Gulf was found to have a low ATWS frequency (see Appendix A) other Mark III plants may not have all of the features which contribute to its low frequency.

Function: Operator Response During ATWS

Guideline 2.A. Operator Response During ATWS

Basis: 2.A. Significant study and research have preceded the current work on severe accidents, in particular, reference is made to the rulemaking activity already accomplished on the ATWS subject. The criteria developed here are based on the assumption that each of the plants are (or will be) in compliance with the ATWS Final Rule dated July 26, 1984. In addition, the following reflects specific measures that complement and supplement the ATWS Rule particularly in the area of the operator's role and function.

During an ATWS sequence the operator is required to inhibit initiation of automatic safety systems and attempt to manually control and mitigate the outcome of the event. In contrast, most other accident sequences are prevented or mitigated by systems which allow the operator to monitor automatic system initiation and require intervention only when a system fails to function adequately. Thus, an ATWS sequence requires operator responses which are in opposition to the highly trained responses required for the recovery and mitigation of all other off-normal and accident events. Therefore, operator training and procedures for the ATWS sequences must specifically prepare operators to perform the contradictory actions as well as the other required measures below:

Criteria:

2.A.1. Operator training and procedures should specify the plant parameters that are indicative of ATWS and the actions to be taken to verify that the reactor recirculating pumps have tripped automatically. Additionally, they should specify the actions to be taken if automatic trip of the reactor recirculating pumps does not occur.

2.A.2. Operator training and procedures should ensure reactor water level control during ATWS, specifically, keeping water level at top of active fuel. Note: This unique control requires actions which conflict with mitigating actions for all other accidents, which call for flooding the reactor core.

Table 4.2 (Cont'd) Proposed Criteria for BWR Mark III Containment Preliminary Guideline 2: Mitigation of Anticipated Transients Without Scram (ATWS) Sequences

2.A.3. Operator training and procedures should specify that when reactor water level approaches the top of active fuel during an ATWS, these level indicators may be inaccurate.

2.A.4. The Automatic Depressurization System (ADS) should be capable of being reliably defeated by the operator prior to its automatic initiation. Operator training and procedures should address the possible reluctance of operators to defeat a safety system, in particular, the need to inhibit the ADS immediately after SLCS initiation attempt.

2.A.5. Operator training and procedures should specify the responsibilities of operating staff crew members and clarify how information will be exchanged among them. In particular, instrumentation readings may have to be relayed between the crew member(s) operating the control boards and the senior reactor operator coordinating the crew's response to the accident.

2.A.6. Operator training and procedures should specify the plant parameters indicative of automatic SLCS actuation and the actions to be taken to verify that SLCS was actuated. If not automatically actuated, they should specify the actions and conditions for manual SLCS initiation.

2.A.7. The systems and equipment required to be interfaced by the operator as specified by this guideline should be designed to perform their function in a reliable manner accounting for the predicted environmental and fluid dynamic loads.

Guideline 2.B. Plants with Automatic Initiation of a Two Train Standby Liquid Control System (SLCS)

Basis: 2.B. PRAs which have investigated plants with automatic initiation of two train SLCS, alternate rod insertion and high capacity SLC boron injection systems have found greatly reduced ATWS frequency (less than 10^{-7} per year).

Criterion:

2.B.1. For those plants that have a reliable two train SLCS with automatic initiation and are in full compliance with the ATWS Final Rule dated July 26, 1984, no additional criteria are necessary.

Table 4.3 Proposed Criteria for BWR Mark III Containment
Preliminary Guideline 3: Mitigation of Station
Blackout Sequences

Concern: Station blackout sequences have been shown to be one of the leading classes of severe accident sequences both in terms of core damage frequency and risk.

Functions: RPV Injection (Guideline 3.A)
Hydrogen Control (Guideline 3.B)

Guideline 3.A. RPV Injection

Basis: 3.A. Significant study and research have preceded current work on severe accidents, in particular, reference is made to the rulemaking activity already under way on Station Blackout. Nevertheless the following reflects specific measures that complement and supplement the proposed Station Blackout Rule particularly in the area of decreasing the core melt frequency during station blackout conditions by improved RPV injection. It is assumed that the plants will be in full compliance with the proposed Station Blackout Rule.

Criteria:

3.A.1. For improved long-term RPV injection capability, either item a) or b) below should be implemented:

a) For the BWR-6 design, the Reactor Core Isolation Cooling (RCIC) System is intended for the purpose of RPV injection independent of AC power. However, it has been demonstrated that this system cannot sustain itself in the presence of a prolonged blackout. Therefore, the following should be incorporated into this design; namely, the RCIC should be capable of performing its intended function in the presence of station blackout conditions.

b) Given that the HPCS diesel is not part of the station blackout, the HPCS system should be capable of performing its intended function while under station blackout conditions.

Guidance:

The intended long-term injection system a) or b) above should have:

(i) A dedicated DC control system which will survive for an extended period without room cooling.

(ii) Alternative water supplies to provide makeup for decay heat removal after the suppression pool becomes too hot to meet pump design criteria.

Table 4.3 (Cont'd) Proposed Criteria for BWR Mark III Containment Preliminary Guideline 3: Mitigation of Station Blackout Sequences

(iii) The turbine or diesel and associated pumps should have sufficient self cooling to operate for an extended period without service water, component cooling water and room cooling.

3.A.2. Operator training and procedures should specify the plant parameters indicative of HPCS and RCIC initiation. Additionally, the training and procedures should specify the actions required to place and/or assure that these systems are in operation under station blackout conditions.

3.A.3. The HPCS and RCIC should be designed to perform their functions in a reliable manner under the predicted environmental and fluid dynamic loads associated with station blackout conditions.

Guideline 3.B. Hydrogen Control

Basis: 3.B. The ASEP study for Grand Gulf[8] found the dominant contributor to core melt to be station blackout. For this type of sequence sufficient hydrogen will be produced to threaten the containment integrity due to hydrogen explosions. However, the present hydrogen control system is AC dependent and will not be available.

Criteria:

3.B.1. Operator training and procedures should specify methods and actions to prevent initiation of the hydrogen control system under conditions which may lead to a hydrogen explosion.

3.B.2. The hydrogen control system should be capable of performing its intended function under station blackout conditions.

Guidance:

A suitable hydrogen control system would have a dedicated power supply system to preserve function for the anticipated hydrogen generation phase of a severe accident resulting from station blackout.

3.B.3. The hydrogen control system should be designed to perform its function in a reliable manner under the predicted environmental and fluid dynamic loads associated with station blackout conditions.

Table 4.4 Proposed Criteria for BWR Mark III Containment Preliminary Guideline 4: Mitigation of Loss of Containment Heat Removal Sequences

Concern: Failure to remove the decay heat buildup in the suppression pool (loss of containment heat removal) following a transient event has been shown to create NPSH problems for the pumps taking suction from the suppression pool and therefore, can lead to injection failure, subsequent core damage, and containment failure. WASH-1400 indicated that this was a leading class of core damage sequences.

Function: Emergency Core Cooling (ECC) Injection - (Guideline 4)

Guideline 4. Emergency Core Cooling Injection

Basis: 4. Implementation of the following criteria will significantly reduce the failure potential of ECC injection due to loss of the containment heat removal function.

Criteria:

4.1. Operator training and procedures should specify methods and actions for heat removal via specified alternate injection path(s) for severe accident conditions when suppression pool temperature precludes use of primary (ECC) injection paths.

4.2. For the alternative injection path(s), it should be demonstrated that the flow be sufficient to preclude core damage for loss of containment heat removal events.

4.3. If local operation of the equipment is required, the time required to perform these functions should be consistent with the time available to help prevent core damage and account for personnel exposure to the predicted severe accident environment.

4.4. Equipment designated for the control and operation of suppression pool cooling should be capable of performing their function in a reliable manner under the predicted environmental and fluid dynamic loads.

Table 4.5 Proposed Criteria for BWR Mark III Containment Preliminary Guideline 5: Analysis of Support System Interdependencies

Concern: When conducting a PRA, IPE or similar analysis, it is imperative that the support system interdependencies be fully developed, understood and reflected in the final results. Otherwise there is no assurance that the dominant core damage/risk sequences have been identified.

Function: Analysis of Support System Interdependencies (Guideline 5)

Guideline 5. Support System Interdependencies

Basis: 5. Implementation of the following criteria will help to assure that the full set of support system interdependencies have been identified and have been reflected in the results.

Note: The following criteria are easily outlined but are not easily implemented. The complex nature of a nuclear power plant makes it imperative that this area of analysis be fully examined. However, as no two plants are exactly the same, especially in the area of support systems, this analysis should be done on a plant-by-plant basis.

Criteria:

5.1. All systems that provide any direct support to either a frontline or support system should be identified along with its supported system. For each dependency that is identified, the failure mechanism and time should be estimated.

5.2. Each dependency should be conditioned as appropriate as to what sequences or under what (if not all) circumstances it applies. In view of its importance, a separate station blackout dependency table should be provided which gives the available systems, their anticipated survival period and the ultimate cause (e.g., no room cooling) of their failure.

5.3. The dependencies should then be linked together (preferably by computer) within the analysis in order that the extent to which their influence reaches through the systems to a consequence will be discovered.

Guidance:

To illustrate this point further, reference is made to the BNL study of system interactions (support system interdependencies) at Indian Point Unit 3.[7] The major finding of that study was that a specific single station emergency battery could fail and among other things, negate the entire low pressure injection function. The point to be emphasized here is that none of the numerous other studies and reviews of the Indian Point 3 design were able to detect this important single failure nor did the BNL study until all of the support systems were explicitly modeled, linked together and solved using the SETS computer code.

Table 4.6 Proposed Criteria for BWR Mark III Containment Preliminary Guideline 6: Maintenance of Containment Integrity

Concern: Breach of the containment boundary in the progression of a severe accident can lead to significant releases of radioactivity.

Functions: Wetwell Venting of Noncondensible Gases - (Guideline 6.A)

<u>Guideline 6. Provide Wetwell Venting</u>

Basis: Implementation of the wetwell venting will significantly reduce the potential for loss of containment integrity due to overpressurization events.

Caution: Containment venting should not be indiscriminantly performed. A clear understanding of the accident sequence in progress should have been assessed prior to initiating venting. The effects of venting should have been assessed and made known to the operators during the training program. The assessment should include the effects of containment venting on the operation of ECC injection systems and health consequences.

Criteria:

The following should be assessed to ensure wetwell venting capability:

6.1. For accident sequences where wetwell venting has been assessed to be beneficial, wetwell venting should commence, except for a station blackout, when containment pressure reaches the predetermined containment venting pressure set point. In selecting the containment venting pressure set point, the following functions should be assured:

a. the ultimate containment pressure capability would not be exceeded,

b. the backpressure acting on the safety relief valve assemblies would not prevent them from performing their function, and

c. the vent valve assemblies would not be prevented from performing their function.

During a station blackout, wetwell venting should commence in accordance with the criteria developed using the BWR Emergency Procedure Guidelines, i.e., following the onset of the transient (before depletion of the station batteries). If station batteries are not available, manual initiation of wetwell venting is required (see Criterion 6.2).

6.2. If manual initiation of wetwell venting is required, the time required to perform this function should be taken into account in the training and procedures to preclude the potential for exposing personnel to the harsh environment. Otherwise, the containment venting valve should be powered from a source independent from the plant emergency power sources.

Table 4.6 (Cont'd) Proposed Criteria for BWR Mark III Containment Preliminary Guideline 6: Maintenance of Containment Integrity

6.3. Operator training and emergency procedures should specify the plant parameters that will prompt the operators to make preparation, commence and terminate the venting sequence. The training and procedures should also be consistent with the required actions and timing of those actions so venting will commence immediately when required (see Criteria 6.1 and 6.2). The training and procedures should further specify the flow path(s) available for venting, specific components to be aligned, and the required positions/states for these components. The training and procedures should specify how to proceed if termination of venting is not possible especially as they relate to emergency management.

6.4. Venting capacity should be greater than the predicted rate of increase of the containment pressure during sequences where venting is anticipated and meet the requirements of Criterion 6.1.

6.5. The criteria for filtering is dependent on the potential for bypassing the suppression pool. Whether the suppression pool is bypassed or not, the radiological release should be reduced by an order of magnitude compared to no filtering. Except for that portion of the containment atmosphere that is present prior to initiation of the severe accident, the venting flow path should ensure that all media to be vented passes through the suppression pool thus providing filtering by the pool.

6.6. Equipment designated or used to support wetwell venting should be capable of performing that function in a reliable manner for a sufficient period to include vaporization release phase of core concrete interaction under the predicted environmental and fluid loads associated with the venting commencement pressure (see Criterion 6.1).

6.7. Support systems for the venting valves (electrical power for a motor-operated valve, air and/or spring/piston for an air-operated valve) should be of sufficient capacity that at minimum power or pressure provided, the delivered force from the actuator is greater than the resulting fluid loads for all angles of opening.

6.8. The effects of possible hydrogen burn, radiation or steam on equipment located in the reactor building outside of the primary containment should be considered for venting through all possible flow paths. If equipment important to the mitigation of accident sequences are jeopardized by venting, alternate venting paths should be identified, assessed, and judged not to be detrimental for venting.

6.9. The effects of possible containment depressurization on the emergency core cooling pump net positive suction head should be assessed. Alternate injection sources which are unaffected by venting should be identified.

Table 4.6 (Cont'd) Proposed Criteria for BWR Mark III Containment Preliminary Guideline 6: Maintenance of Containment Integrity

6.10. Wetwell venting should be terminated when significant radioactive noble gas inventory begins to appear in the wetwell air space such that the projected offsite releases approaches a level that could be life threatening.

Appendix A

SEVERE ACCIDENT RISK INSIGHTS

This section considers various studies of BWR-6 reactors with Mark III containments. The paradigm chosen for the present purposes is the Grand Gulf design. Insights from selected studies that contributed to the development of the specific preliminary guidelines (Section 3) for the prevention and mitigation of severe accidents are identified and discussed.

The approach used here, to characterize the BWR-6, Mark III risk profile, employs the Peach Bottom (BWR-4, Mark I) analysis[1] as a stepping stone. As Peach Bottom was the first plant analyzed in this severe accident program, an extensive analysis and comparison of the various studies was performed and documented not only to identify the important areas of risk for Peach Bottom but also to form the baseline analysis for the plant analyses that followed. After comparison of the various BWR-4-related studies, it was concluded in that analysis that all of the studies pointed to the same key features as being important. Based on this insight, the focus of this effort was to determine the dominant sequence types for Mark III plants by applying a set of screening criteria given in Section 3 and to identify the associated containment failure modes.

A.1 Core Damage Profile

The core damage profiles from a number of BWR-6 plant studies have been compiled in Table A.1. These studies include the Grand Gulf RSSMAP[2] and the Grand Gulf IDCOR[3] analyses and the results found in the BNL Review[4] of the GESSAR II PRA.[5] Explicit references to the GESSAR II work itself have been avoided due to its proprietary nature.

A second set of core damage profiles relating specifically to Grand Gulf are found in Table A.2. This table was constructed in the attempt to reconcile the formatted presentation of the RSSMAP and IDCOR results from Table A.1 to the format and content of the (draft) ASEP[6] study of Grand Gulf. This reconciliation process was performed for the following reasons: a) ASEP explicitly presented station blackout sequences and these overwhelmingly dominate the estimated CDF whereas the two other studies have station blackout cut sets within other sequences and, b) the ASEP results were not grouped by initiator into the two categories of T_1 (loss of offsite power) and T_{23} (all other transients) as were the other two studies.

For the RSSMAP study, all of the leading cut sets presented for the T_1 initiator in the report[2] were examined to identify the contribution of station blackout conditions (i.e., loss of both offsite power and diesel generators 1 and 2). All cut sets displaying these conditions were subtracted from their original sequences and added together to form a station blackout sequence. The remaining (non-blackout) sequence contributions (T_{23}) were then combined with the remaining T_1 contributions and presented in Table A.2.

Leading sequence cut sets were not provided in the IDCOR analysis. However, Figure A-18[3] does provide the station blackout contribution within the T_1QUV sequence. Therefore, T_1QUV was reduced by the station blackout (TB) portion and combined with T_{23}QUV in Table A.2. As no other station blackout contributions were identified, the remaining like sequences were combined (T_1 + T_{23}) and presented in Table A.2 along with the derived TB contribution.

The following subsections will address the leading CDF sequences including appropriate comparisons between the referenced studies.

Station Blackout

From Table A.2 it can be seen that station blackout is relatively significant in all three Grand Gulf studies. Station blackout is defined as loss of offsite power coupled with failure of the Train 1 and Train 2 emergency power systems. Loss of the Train 3 (HPCS) diesel generator/emergency power system is not required in order to assure a core melt assuming no recovery.

The ASEP study focused directly upon station blackout and modeled station blackout explicitly within the event trees. Figure A.1 presents the five leading blackout sequences which totally dominate the calculated ASEP CDF. From Figure A.1 it can be seen that dependent failure of the HPCS (U_1) and

RCIC (U_2) systems are key contributors to these sequences. Both short-term sequences (3 and 5) include the independent failures of both HPCS and RCIC, and the long-term sequences show RCIC or HPCS as an initial success and yet all five sequences shown in Figure A.1 result in core damage and containment failure. This is due to the modeling assumptions applied to the HPCS and RCIC systems. The HPCS and RCIC systems have been modeled with three time-dependent failure modes in the blackout sequences. These are failures due to 1) loss of dc control power due to battery depletion in 11-12 hours (this only applies to HPCS if its dedicated diesel generator has also failed), 2) pump seal failure in a 6-8 hour period based upon temperature rise in the suppression pool, and 3) failure due to loss of room cooling in about twelve hours. Due to the shorter time to failure, pump seal failure is dominant.

Table A.3 shows the possible effects of removing these time-dependent blackout sequence failure modes of HPCS and RCIC. The results of this assumption show that short-term failures would then represent the blackout core damage frequency contribution and the overall total core damage frequency contribution would be reduced, in this example, by almost an order of magnitude.

In the IDCOR analysis, the HPCS and RCIC systems are not modeled as guaranteed failures in the long term, given failure to recover an ac power source, as was discussed above. Rather, the IDCOR analysis assumes that the batteries will deplete in about five hours (versus the 11-12 hours in ASEP) and this failure will therefore occur before the seal failure assumed in ASEP. Battery depletion therefore turns out to be the more benign dependent failure mode as it leaves the HPCS and RCIC systems (pump seals) undamaged and available for subsequent recovery of ac power. In summary, the assumed time to battery depletion is the overwhelming driving factor between the difference in quantification between IDCOR and ASEP. It is clear that no matter which of these two studies more accurately reflects the present Grand Gulf plant, the key to reducing the station blackout contribution to core damage frequency is not in simply adding battery capacity, but rather in providing long term protection to the pump seals.

In addition, the results of both the GESSAR II PRA[5] and the BNL review[4] of that document are essentially in agreement with the Grand Gulf ASEP study in that LOOP (predominantly station blackout) events are the dominant contributors to core damage.

Anticipated Transient Without Scram (ATWS)

The core damage frequency resulting from an ATWS event has been significantly reduced from the RSSMAP study to the IDCOR "committed" results and even further in the ASEP results (Table A.2). The contributing factors to this are as follows. First, within the IDCOR analysis, the addition of an Alternate Rod Insertion (ARI) system has effectively decreased the scram failure frequency by a factor of three. Another factor is the doubling (from 43 to 86 gpm equivalent) of the boron flow of the SLCS. This has the effect of allowing more time to elapse before SLCS must be activated. SLCS failure in the RSSMAP study was dominated by operator failure to actuate. Therefore lower operator failure probabilities are used in the IDCOR analysis to reflect the additional time available. Additionally, there now exist emergency procedure guidelines that facilitate the operator aling with an ATWS event including the possibility of depressurizing and using low pressure injection systems.

The ASEP ATWS analysis does not include any additional hardware upgrades beyond that of the IDCOR analysis. It does however, appear that a more detailed analysis has been performed of the probability of operator error and of the low pressure injection procedures, these two areas seem to account for the further reduction in calculated ATWS CDF. Specifically, a more rigorous look at operator actions given detailed ATWS procedures and training has yielded lower HEPs in the ASEP analysis. Also, the resulting lower HEP asociated with actuating both trains of SLCS within ten minutes is assumed to keep the suppression pool temperature at or below 180°F. This results in a lower probability of failure of the HPCS pump seals, thus reducing the probability of one of the failure modes for high pressure injection.

The ATWS rule assumes that BWR-6 desigrs which incorporate the hardware features noted above need not make additional significant hardware modification to keep ATWS core damage probabilities low. As can be seen from the results of the ASEP and IDCOR analyses, further reduction in CDF is still possible by upgrading operator performance. Based upon this result, operator actions in response to an ATWS event are addressed in one of the proposed guidelines to assure that other BWR-6's do not have a high ATWS CDF. It is further noted that the GESSAR II PRA Review had comparable results to the IDCOR analysis (See Table A.1).

Loss of Containment Heat Removal

In the RSSMAP study, the phenomenon of loss of containment heat removal dominated the CDF. In the later studies, these types of sequences no longer make a significant contribution. The RSSMAP analysis included the assumption that containment failure resulted in loss of injection capability. According to the ASEP study: "It has been determined that deformation of injection lines does not occur, and since the systems that take suction from the suppression pool can pump saturated water, loss of injection does not occur as a result of containment failure."

In addition to the change in the assumption of injection failure noted above, the IDOCR and ASEP studies also investigated ways of delaying or preventing containment failure itself which were not accounted for in the RSSMAP study. These mechanisms include venting the containment and the use of alternate injection paths. When these alternate success criteria are factored into the Grand Gulf model, as was done in the IDCOR and ASEP analyses, the estimated CDF for loss of containment heat removal sequences is reduced to less than 10^{-7} per reactor year. In accordance with these insights, a guideline has been proposed in Section 4 to assure that other BWR-6 Mark III designs incorporate emergency procedures that will ensure that loss of containment heat removal sequences are kept at a similarly low level.

Interfacing Systems LOCA

Traditionally, interfacing system LOCAs have been identified in PRAs as low frequency but high consequence events. The Grand Gulf studies do not identify these sequences as high risk events. It appears that the basis for this result stems from the very low to negligible probabilities calculated for these events. BNL is currently conducting a study of BWR/PWR interfacing system LOCAs (Generic Issue 105). It is recommended that this issue not be dropped from the severe accident guideline list (as would be suggested by the results of the Grand Gulf studies) pending resolution of this generic issue.

A.2 Core Meltdown Phenomena and Containment Response

In the previous section important core meltdown accident sequences were identified in terms of the overall core melt frequency. In this section, a review of the core meltdown phenomena and containment response appropriate to these accident sequences is presented. In addition, accident sequences are examined which, although they do not appear to be important to the overall core melt frequency, may pose a unique or very severe threat to containment integrity. We will again rely heavily on the IDCOR and SARP[6] analyses which were specifically carried out for the Grand Gulf plant. We also will take into account other studies pertinent to a BWR-6 with a Mark III containment and, in particular, the Containment Loads Working Group (CLWG) report[7] and the Containment Performance Working Group (CPWG) report.[8]

A typical Mark III containment building is shown in Figure A.2. The Mark III containment relies on water to condense any steam that might be released from the primary system during an accident. Containments of this design are called pressure suppression containments. Mark III containments are very effective at condensing steam but they may be vulnerable to buildup of combustible and noncondensible gases that would be generated during a severe core meltdown accident.

The aim of this section is to identify severe accident threats to the containment appropriate to the accident sequences identified in Section A.1. These threats are then used to determine the most probable mode of containment failure. This, in turn, identifies the potential release paths for fission products to reach the environment. This section, therefore, provides the link between the identification of core meltdown accident sequences and the determination of fission product release paths.

Reference 9 is a Battelle Columbus study of the Mark III containment responses to the two leading core damage sequences, i.e., blackout and ATWS.

The blackout sequences have been divided into two scenarios and the ATWS sequences are grouped into one scenario.

One blackout scenario assumes that containment failure follows soon after vessel failure due to a hydrogen explosion. The other blackout scenario assumes that a hydrogen explosion does not occur either because of a slow burning rate or lack of ignition source. The long term failure of the containment then occurs as the result of overpressurization due to noncondensible gas generation. In both scenarios it is assumed that there is leakage that bypasses the pool. The amount of this leakage has a direct bearing on the magnitude of the radioactivity released and is assumed to be much greater for the short term hydrogen explosion scenario.

The ATWS sequences are characterized by containment overpressure failure prior to core damage. The containment is overpressurized due to power generation greater than that which can be removed by the residual heat removal system. This causes the suppression pool to heat up and pressurize the containment to failure in a rather rapid fashion. Loss of the containment is assumed to cause loss of the emergency core cooling systems and thus core damage.

The Grand Gulf studies also point to a possible suppression pool bypass mechanism in which corium ejected from the vessel may erode through the pedestal wall causing displacement of the vessel which in turn disrupts penetrations through the drywell wall with the possibility of pool bypass. The IDCOR study investigated the addition of a drywell spray system to prevent this situation. The IDCOR analysis stated that a drywell spray system can be used to reduce drywell pressure, cool the drywell, and quench the melt and reduce radioactivity in the drywell in case of a core melt accident. It further stated that the system could preserve containment integrity during an accident by quenching the corium and reducing the likelihood, size and consequences of a potential release. The IDCOR analysis concluded, however, that this modification was not necessary since the probability of pool bypass sequences was estimated to be low even without the drywell spray system.

The results of an assessment of core meltdown phenomena and containment response is usually expressed in terms of a containment matrix. A containment matrix provides the framework for estimating the conditional probabilities of a particular accident sequence resulting in a variety of containment failure modes (or fission product release paths).

The IDCOR and SNL SARRP containment matrices are given in Table A.4. From an inspection of Table A.4 it is clear that the SARRP approach includes a higher potential for drywell leakage and pool bypass than the IDCOR approach. Differences in the probabilities in Table A.4 are due to differences in modeling assumptions for core meltdown and containment response in the IDCOR and SARRP studies. These differences are discussed in detail in the following sections.

The review of the IDCOR and SARRP analyses of core meltdown phenomena and containment response was greatly assisted by the IDCOR/NRC meetings that have been held specifically to identify differences between the approaches adopted by the two groups and to develop a way of resolving these differences. These meetings identified eighteen broad NRC/IDCOR issues that highlight significant differences between the approaches of the two groups. These issues are listed in Table A.5 but they do not all apply to a BWR and some are not related to core meltdown phenomena and containment response. Of the eighteen issues, eight have been identified that are pertinent to the subject of this section. Each issue is discussed in turn in the following sections. Differences between IDCOR and SARP will be identified and their significance indicated.

A.2.1 Invessel H_2 Generation (NRC/IDCOR Issue 5)

There are significant differences between the IDCOR and SARP predictions of H_2 generation during invessel core melting. During the early stages of core heatup and degradation (while the fuel rods are still in place in the core region), both IDCOR and BCL predict similar H_2 generation. However, after the fuel rods and cladding begin to melt and relocate into the bottom of the reactor vessel, the BCL analysis with STCP indicates more H_2 generation than the IDCOR analysis.

Hydrogen is important to containment loading because it is a combustible and noncondensible gas. The Mark III containments are not inerted but are required to have igniters installed to burn any H_2 that may be released in a controlled manner. However, H_2 combustion (Issue 17) could be a threat to the Mark III containments if the igniters fail to burn the hydrogen gradually. For instance, recovery of AC power after initial blackout could lead to activation of the ignitors with dangerous levels of hydrogen existing in the containment, according to the SNL analysis. Mark III containments are also susceptible to the long term buildup of noncondensible gases (such as H_2 and CO_2) which could threaten containment integrity by overpressure. The larger amount of H_2 generated invessel in the BCL analysis leads to a higher predicted containment pressure prior to vessel failure than in the IDCOR analysis. BNL staff have performed an extensive assessment[10] of invessel H_2 generation particularly with regard to accidents that resulted in core damage but which were terminated by subsequent coolant injection. The results of these calculations indicate the potential for more H_2 generation than predicted by IDCOR.

The differences in H_2 generation were found[1] to have very little impact on risk for the Mark I containment since it is inerted. However, the Mark III containment is not inerted and the difference in hydrogen generation appears to have an important effect on risk. This is doubly so since the only hydrogen control device (igniters) will not function during one of the dominant core melt sequences (station blackout).

A.2.2 Core Slump, Core Collapse, and Reactor Vessel Failure (NRC/IDCOR Issue 6)

This is another area where there are significant differences between the IDCOR and ASEP/SARRP analyses. The importance of these differences to overall risk again depends on plant specific systems. Section A.2.1 indicated that the predicted hydrogen generation during core slump was quite different in the IDCOR and BCL analyses. The larger hydrogen generation contributes to a larger probability of significant drywell to wetwell leakage (.42) for SARRP (refer to Table A.4).

The core slump and reactor vessel failure models also significantly influence the initial conditions for exvessel interactions of the core debris with water or concrete. The IDCOR core slump model assumes that after 20% of the core has melted, it will relocate into the bottom of the reactor vessel, which will then rapidly fail due to local penetration failure. Thus only a relatively small fraction of the core will be initially released from the reactor vessel. The remainder of the core melts down over a much longer time period. A similar philosophy has been adopted in the draft NRC staff issue paper on direct heating. This work states that the BWR core support design (which provides individual support for each group of four fuel bundles from the vessel bottom head) is judged to minimize the probability of high pressure ejection of core debris into the containment. Slumping of relative small quantities of core debris (due to localized failure of the supports) is anticipated to result in depressurization of the vessel (due to local melt-through) before large quantities of molten core material have collected in the bottom head.

On the other hand, the STCP core slump model used in the BCL analysis assumes total collapse of the core into the bottom of the reactor vessel after 75% of the core is predicted to melt. Thus, a large fraction of the core debris is available to be released when the vessel is predicted to fail in the BCL analysis. The much larger quantity of core materials released from the vessel at the time of vessel failure in the BCL analysis has important implications for the Mark III containment. If the primary system is at high pressure during core meltdown, then the molten core materials will be ejected under pressure from the reactor vessel when it fails. In Section A.2.4 the phenomena that could occur when molten core debris are ejected from the reactor vessel under pressure is discussed. Since more core debris is predicted to be ejected, the resulting pressure/temperature loads in containment will be correspondingly higher.

SNL has also performed an uncertainty study in support of NUREG-1150 which examines the range of possible core slump behavior and attaches a low likelihood to the high core-melt fraction slump model.

If the primary system is depressurized during core meltdown, then the core debris will fall into the region below the reactor vessel after it fails. Obviously, if more core debris is predicted to fall into the pedestal region, then the resulting molten pool will be deeper and there will be a greater potential for the core to erode the support pedestal and possibly fail the drywell wall. SNL has identified this as a mechanism for pool bypass after vessel failure with a conditional probability of approximately .02.

From the above discussion, it is clear that differences between the IDCOR and BCL/SNL analyses for core slump and vessel failure are significant. The potential for early containment failure depicted in IDCOR and SNL containment event trees (refer to Table A.4) is in substantial agreement in spite of these differences. However, the effect of phenomenological uncertainties has not been addressed yet. The authors concur with IDCOR and SNL in attaching a small probability to early failure of the drywell wall due to contact with core debris which would result in large releases.

A.2.3 Containment Failure Due to Invessel Steam Explosions (Issue 7)

The potential for an invessel steam explosion to occur and generate a missile capable of failing containment was investigated by a group of experts and the results published in NUREG-1116.[11] The conclusion of this expert group was that such an event has a relatively low probability. These results are reflected in the SNL and IDCOR containment event trees. The allocation of a very low conditional probability (10^{-4} per reactor year) of occurrence to this event is supported by the authors.

A.2.4 Direct Heating of Containment (Issue 8)

This is an area of significant phenomenological uncertainty related specifically to core meltdown with the primary system at high pressure. If molten core materials are ejected from the reactor vessel under pressure, experiments[12] at SNL have indicated that they form fine aerosols, which are dispersed into the containment atmosphere and directly heat it. An additional concern is the oxidation of the metallic content of the core debris. These reactions are very exothermic and would add an additional heat load to the containment.

The pressure rise in containment due to direct heating is directly proportional to the quantity of core debris dispersed from the reactor vessel.

Section A.2.2 noted that the BCL analysis predicts significantly more debris release at vessel failure than the IDCOR analysis. Thus the potential for early containment failure due to direct heating is higher in the BCL analysis. However, the SNL event trees attach a high probability to a slow melt release from the vessel and thus failure due to direct heating is low.

The assumption that all the core debris is released at vessel failure (BCL analysis) is clearly conservative. The IDCOR and SNL analyses appear to be too optimistic considering the lack of supporting large scale experiments. In addition to the pressure loads imposed by the dispersed core materials, there is the concern that the hot core debris could erode the support pedestal and fail it (see Section A.2.2).

A.2.5 Exvessel Heat Transfer Model from Molten Core to Concrete (Issue 10)

This issue is of concern to Mark III containments because heat transfer from the top of molten core materials (on the drywell floor) directly heats the drywell atmosphere. Thus, differences in heat transfer from the top of the core debris can result in significant differences in the predicted drywell atmospheric pressures and temperatures. The IDCOR model[13] transfers more heat from the top of the core debris than the STCP model (CORCON Mod 2[14]). Thus, IDCOR predicts higher drywell temperatures than the BCL analyses. However, because IDCOR predicts high heat transfer from the top of the core debris, the concrete erosion velocities are much lower than the BCL predictions. Lower concrete erosion results in less gases and aerosols released from core/concrete attack and thus lower pressures in containment.

Differences in the predicted drywell pressure/temperature histories can influence the potential for suppression pool bypass (Issue 13A) and containment performance (Issue 15).

A.2.6 Suppression Pool Bypass (Issue 13A)

If the fission products pass through the suppression pool both IDCOR and BCL predict significant retention of fission product aerosols in the water. The amount of retention depends on several factors such as submergence, water temperature, aerosol particle size, carrier gas composition, and others. The ability of the Mark III suppression pool to trap aerosol fission products is an important mitigative feature. Thus, any pathways that might open, which would allow the fission products to bypass the pool are very undesirable. The following are possible ways in which the suppression pool may be bypassed:

- failure of the drywell wall due to hydrogen explosions
- failure of vacuum breakers between the drywell and wetwell
- failure of drywell penetrations due to high temperature
- failure of the pedestal wall as a result of contact with molten core materials.

Because of the importance of the suppression pool as a mitigative feature, the vulnerability of a Mark III containment to any of the above bypass pathways must be carefully assessed. The probability of degradation of the drywell penetrations due to high temperatures in the SNL analysis reflects much of the work of the CPWG, which had significant BNL input. Failure of the pedestal wall as a result of contact with molten core materials and the resulting displacement of the vessel is an area of great phenomenological uncertainty. Preliminary event trees from SNL indicate that substantial leakage of approximately 1 square foot through the drywell wall will occur after failure of the pedestal. We are unable to rule out pedestal failure as a potential cause of pool bypass, but the capability of the upper pool to dump into the drywell and quench the core appears to be an important mitigative capability in Grand Gulf.

A.2.7 Containment Performance (Issue 15)

The response of a Mark III containment to severe accident loads is uncertain. In Section A.2.5, we noted that IDCOR predicts very high drywell temperatures but does not predict drywell failure. IDCOR assumes that a relatively small opening will occur in the outer containment which allows gradual leakage with no pool bypass. By comparison the BCL analysis with the STCP allows for primary containment failure due to overpressure and assumes an opening large enough to rapidly depressurize the primary containment. In addition, the BCL analysis allows for degradation of drywell seals due to high temperatures. Seal degradation was assumed to result in a gradual leakage from the drywell in the BCL analysis.

Differences in containment performance can influence the timing and quantities of fission products released to the environment (refer to Section A.3). However, these differences do not lead to major differences in the predicted overall risk as discussed in Section A.4.

A.2.8 Hydrogen Ignition and Burning (Issue 17)

Although there are considerable differences in the rates of hydrogen generation both IDCOR and BCL predict that hydrogen burning gives a high probability of early containment failure for station blackout. Although Grand Gulf has installed igniters to prevent hydrogen detonation in compliance with the interim hydrogen rule, the igniters require AC power and are not available for the dominant class of core melt accidents (station blackout) and may actually exacerbate the situation if power is restored later in the accident when detonable levels of hydrogen have accumulated.

A.3 Fission Product Release

Section A.2 identified potential containment failure modes or fission product release paths appropriate to the important core meltdown accident sequences identified in Section A.1. The aim of this section is to determine the timing and amount of fission products released from the damaged fuel and predict the subsequent mitigation of these fission products along the release paths identified in Section A.2. The IDCOR and BCL analyses for the Grand Gulf plant are used as the basis for these calculations.

In order to review the differences in approach, IDCOR and NRC contractor analyses (performed for SARP) are compared in Tables A.6 and A.7 for ATWS and Station Blackout sequences respectively. The IDCOR methods predict lower releases of all fission product groups than predicted by BCL. The reasons for the different predictions in Tables A.6 and A.7 are complex but were discussed during the numerous IDCOR/NRC meetings and they are included in the list of eighteen NRC/IDCOR issues in Table A.5. Out of the eighteen issues, six are pertinent to fission product release and transport. However, not all of the six are major contributors to the differences in Tables A.6 and A.7. The most prominent differences are displayed in Table A.7 for station blackout. The BCL analysis assumes a large leakage through the drywell wall thus bypassing the pool while IDCOR assumes that there is no pool bypass. Each of the six NRC/IDCOR issues is discussed in the following subsections.

A.3.1 Fission Product Release Prior to Vessel Failure (Issue 1)

This is one issue that does not contribute significantly to the differences between the IDCOR and BCL analyses in Tables A.6 and A.7. Both studies predict similar releases of the more volatile fission products during invessel core degradation with the exception of Te. However, a recent report by IDCOR assessed the impact of Te treatment and modeled similar invessel Te releases to the SARP analyses. Differences in the predicted environmental releases of Te in Tables A.6 and A.7 are therefore not due to differences in the invessel Te release and retention models but due to differences in the amount of fission products which are assumed to bypass the pool (refer to Section A.3.6).

A.3.2 Fission Product and Aerosol Retention in the Primary System (Issue 4)

Differences in the initial primary system retention predicted by IDCOR and BCL are again not too significant and differ by less than a factor of two. The important difference between the IDCOR and STCP models is that in the STCP analysis fission products retained in the primary system at the point of vessel failure are permanently retained whereas in the IDCOR analysis revaporization of these fission products after vessel failure is modeled. This is discussed in more detail in Section A.3.4.

A.3.3 Exvessel Fission Product Release (Issue 9)

There are significant differences between the IDCOR and BCL analyses for fission product release as a result of core/concrete interactions. The higher releases of the Sr, La, and Ce groups in Tables A.6 and A.7 in the SARP analyses are due to the modeling of exvessel fission product release. The potential for fission product release and inert aerosol generation during core/concrete interactions was not modeled in the IDCOR analysis of Grand Gulf. IDCOR argued that by modeling the aerosol generation during core/concrete interactions the increased aerosol density in containment would increase aerosol agglomeration and settling, thus reducing the predicted environmental release fractions relative to those predicted without this additional aerosol source.

We do not consider that this IDCOR argument has been adequately supported. In addition, the IDCOR predicted core debris temperatures during core/concrete interactions are very high; based on experimental evidence, one would expect the release of some of the refractory fission product groups at these temperatures. We therefore believe that IDCOR should calculate the release of the refractory fission products and the associated inert aerosols. The BCL analysis currently models the release of the refractory fission products and the inert aerosols and the environmental release fractions are not low (refer to Tables A.6 and A.7).

A.3.4 Revaporization of Fission Products from the Primary System (Issue 11)

Section A.3.2 indicated that revaporization is an area of major difference between the IDCOR and SARP analyses. SARP does not model revaporization of fission products from the primary system after reactor vessel failure whereas IDCOR does model this effect. The IDCOR revaporization model means that significantly more of the volatile fission products are predicted to be released from the primary system later in the accident sequence than in the NRC contractor approach in which revaporization is not modeled. In spite of the high revaporization in the IDCOR analysis, the IDCOR release fraction remains small since they assume no bypass of the pool (Isssue 13A).

A.3.5 Fission Product Deposition Model in Containment (Issue 12)

This is another issue that does not contribute significantly to the differences between the IDCOR and SARP analyses in Tables A.6 and A.7. Issue 13A (suppression pool bypass is discussed in Section A.2.5) really drives the differences in these two analyses. However, this issue may be of more importance to other containment designs.

A.3.6 Secondary Containment Performance (Issue 16)

The Mark III design does not incorporate a secondary containment (reactor building). Therefore, it is not an area of major difference between the IDCOR and SARP analyses.

A.4 Offsite Consequences

In this section the potential offsite consequences of the severe accidents described in the previous sections are examined. There is one NRC/IDCOR issue related to offsite consequences, which concerns differences in the assumed evacuation models. Differences in the evacuation model influence the predicted early health effects. The issue is largely resolved and is related to the fraction of the population assumed not to participate in the evacuation.

Table A.8 gives the consequence calculations for IDCOR and SARP for several accident sequences and failure modes. This table indicates that if the containment is predicted to fail (either early or late) and the suppression pool is not bypassed, then the offsite person-rem predictions are similar ($\sim 10^5$) for the accidents considered. The only time that a significant increase ($\sim 10^7$) in person-rem is calculated is with pool bypass. These results clearly show that preventing of pool bypass is the key to mitigating the fission product releases for a Mark III containment.

A.5 Summary and Risk Insights

A.5.1 Core Damage Profile

Transients dominate the core damage risk profile for the studies examined in Section A.1. For all of the BWR plant PRAs considered, a few sequences figure prominently in all of the respective core damage frequency profiles. This suggests that if the probability of this relatively small subset of accident sequences can be minimized, then there is a reasonable expectation that the overall core damage frequency will be minimized. This principle is used in Sections 3 and 4 to develop preliminary guidelines and proposed criteria to reduce the overall core damage frequency (Goal 3).

It is, however, important to recognize that the qualitative accident sequence descriptors are rather general and broad and that different hardware and/or operational failures in the various BWR-6, Mark III plants could lead to the same general accident sequence. In order to identify the plant specific (and often times unique) potential vulnerabilities that contribute to a given general sequence descriptor (e.g., station blackout) in a given plant, a plant specific examination (such as a failure mode and effects analysis coupled with a fault tree/event tree analysis) is needed.

A.5.2 Consequence Analysis

The assessment of core meltdown phenomena and containment response indicates that the Mark III containment is vulnerable to severe accident containment loads. Unless mitigative actions (specifically H_2 control) are taken, a Mark III containment has the potential to fail within a few hours of vessel failure. The BCL analysis predicts substantial leakage paths from the drywell so that fission products may be released without the benefit of suppression pool scrubbing. The only time that a major reduction in offsite consequences is predicted is if there is no pool bypass. This demonstrates the importance of preventing pool bypass to the mitigation of fission products (Goal 1).

A.6 References

1. W. T. Pratt et al., "Prevention and Mitigation of Severe Accidents in a BWR-4 with a Mark I Containment," Draft BNL Technical Report A-3825R, August 8, 1986.

2. S. W. Hatch et al., "Reactor Safety Study Methodology Applications Program: Grand Gulf #1 BWR Power Plant," NUREG/CR-1659/4 of 4, October 1981.

3. IDCOR Technical Report 21.1, Risk Reduction Potential, June 1985.

4. N. A. Hanan et al., "A Review of BWR/6 Standard Plant Probabilistic Risk Assessment: Volume 1 Internal Events, Core Damage Frequency," NUREG/CR-4135P, May 1985.

5. GESSAR (General Electric Standard Safety Analysis Report) II BWR/6 Nuclear Island, Probabilistic Risk Assessment.

6. A. Kolaczkowski et al., "Reference Plant Accident, Sequence Likelihood Characterization: Grand Gulf," NUREG/CR-4550, Volume 5, (Draft, July 1986).

7. "Containment Loads Working Group Report," NUREG-1079, Draft report for comment, December 1985.

8. "Containment Performance Working Group Report," NUREG-1037, Draft report for comment, May 1985.

9. "Radionuclide Release Calculations for Selected Severe Accident Scenarios BWR Mark III," Battelle Columbus Laboratory, February 1986.

10. J. W. Yang and W. T. Pratt, "Analysis of Hydrogen Production During a BWR-6 Core Heatup Transient," BNL Technical Report A-3808 9-85.

11. "A Review of Current Understanding of the Potential for Containment Failure from In-Vessel Steam Explosions," NUREG-1116.

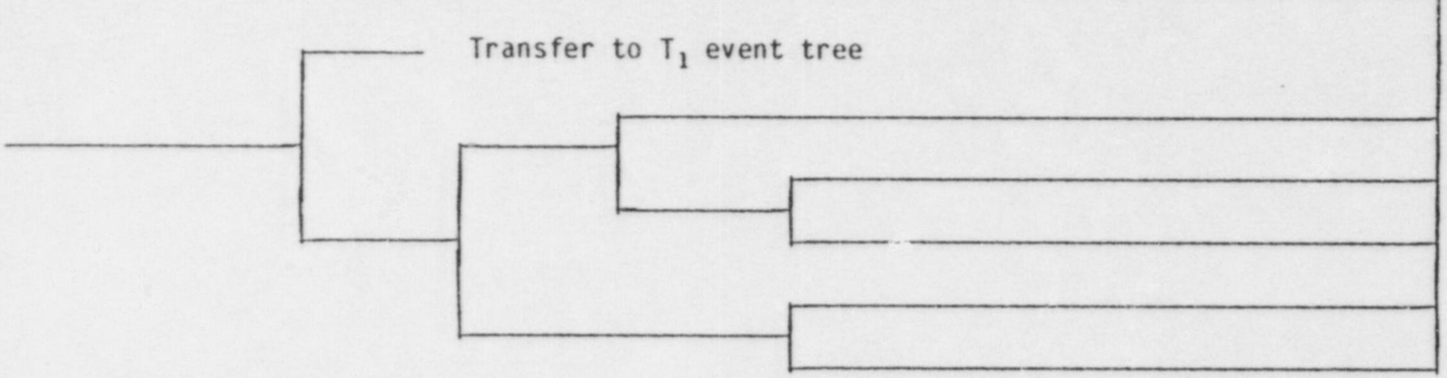
12. Direct Heating Experiments in SURTSEY Facility at SNL.

13. "IDCOR Report Subtask 15 - Molten Core-Concrete Interactions," Atomic Industrial Forum, Technology for Energy Corp., 1983.

14. R. K. Cole, Jr., D. P. Kelly, and M. A. Ellis, "CORCON-Mod 2: A Computer Program for Analysis of Milten Core-Concrete Interactions," NUREG/CR-3920, August 1984.

15. H. Alsmeyer, M. Riemann, and J. P. Hosemann, "Preliminary Results of the KfK MCCI Experiment BETA Facility," 11th NRC Water Reactor Safety Information Meeting, Gaithersburg, MD, October 1984.

Loss of Offsite Power (plus $\overline{RPS}$)	SRVs Reclose (plus SRVs open)	Onsite Power -- DGs A, B Available	Onsite Power -- HPCS DG C Available	High Pressure Core Spray System	Reactor Core Isolation Cooling	RCIC Battery Depletion 11-12 hrs	RCIC Seal/Room Cooling > 12 hrs	Heat Removal (includes venting)	Seq. No.	Core Damage Sequences
T1	P	B1	B2	U1	U2	Z1	Z2	N/Y		



Seq. No.	Core Damage Sequences	
1	$T_1B_1\overline{B}_2\overline{U}_1$	2.1E-5
2	$T_1B_1\overline{B}_2U_1\overline{U}_2$	4.9E-7
3	$T_1B_1\overline{B}_2U_1U_2$	1.5E-7
4	$T_1B_1B_2\overline{U}_2$	2.0E-6
5	$T_1B_1B_2U_2$	2.3E-6

Figure A.1 Grand Gulf station blackout event tree.

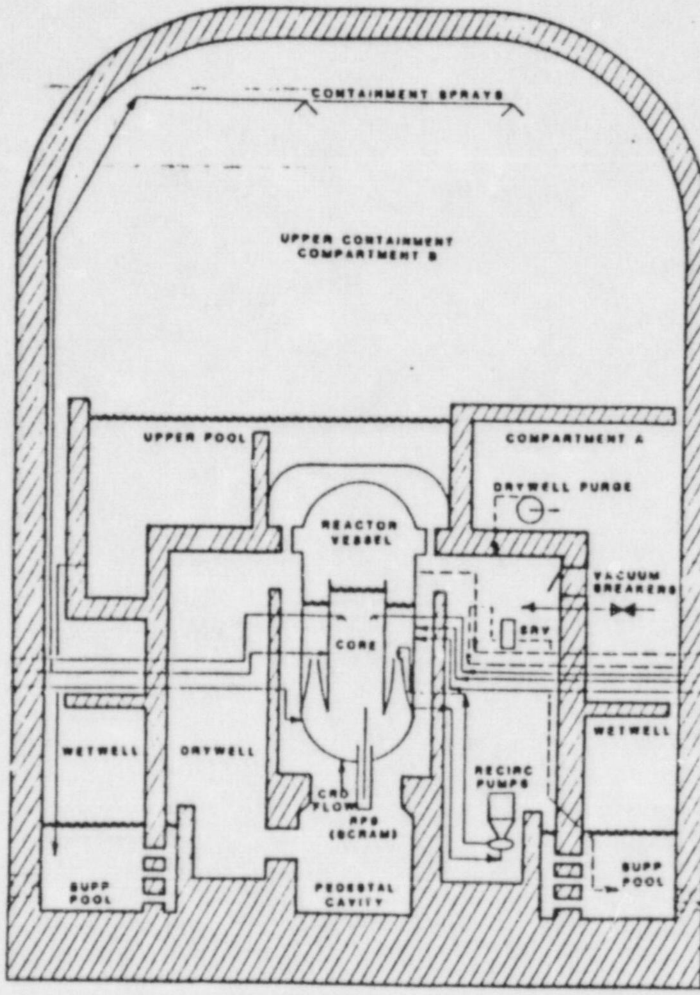


Figure A.2 Mark III containment building.

Table A.1 Selected BWR-6 Core Damage Profiles

Sequence Type	RSSMAP	Grand Gulf IDCOR Committed	GESSAR II PRA Review
$T_{23}QW$	1.2E-5	1.9E-7	1.9E-6
T_1QW	6.2E-6	8.9E-9	--
$T_{23}C$	5.4E-6	1.2E-6	2.52E-6
SI	4.6E-6	5.5E-9	--
$T_{23}PQI$	3.7E-7	2.3E-8	--
T_1PQI	1.6E-6	2.3E-9	--
$T_1QUV + T_1QUX$	1.5E-6	4.7E-7	--
$T_{23}PQE$	5.4E-7	1.7E-9	--
T_1PQE	2.3E-7	--	--
AI	2.6E-7	1.4E-9	--
AE	5.0E-9	1.1E-9	--
$T_{23}QUV + T_{23}QUX$	5.6E-8	1.3E-8	5.3E-7
T_1C	1.2E-7	--	--
T_1QUW	3.4E-8	--	--
$T_{23}QUW$	7.0E-8	--	--
T_1W	--	--	1.6E-6
$T_1UV + T_1UX$	--	--	3.0E-5
$T_{23}UV + T_{23}UX$	--	--	5.3E-7

Table A.2 Grand Gulf Core Damage Profile

Sequence Type	RSSMAP*	IDCOR* Committed	ASEP
TB	1.32E-6	3.4E-7	2.6E-5
TC	5.52E-6	1.2E-6	1.8E-7
TQW	1.71E-5	2.0E-7	--
TPQI	5.18E-6	2.53E-8	--
TQUV	1.42E-6	1.43E-7	3.4E-7
TPQE	7.61E-7	1.7E-9	--
TQUW	1.04E-7	--	--

*These sequence core damage frequencies were derived by extracting explicit station blackout (TB) cut sets and then combining all initiators (i.e., $T_1 + T_{23}$). TB reflects the summation of the extracted cut sets.

Table A.3 Grand Gulf Station Blackout Sequence Core Damage Frequency Point Estimates With Proposed Enhanced HPCS/RCIC Capabilities

		ASEP	Remove or Extend Blackout-Related Time-Dependent Failures[1]
Seq 1	$T_1B_1\overline{B}_2\overline{U}_1$(long term)	2.1E-5	--
Seq 2	$T_1B_1\overline{B}_2U_1\overline{U}_2$(long term)	4.9E-7	--
Seq 3	$T_1B_1\overline{B}_2U_1U_2$(short-term)	1.5E-7	1.5E-7
Seq 4	$T_1B_1B_2\overline{U}_2$(long-term)	2.0E-6	--
Seq 5	$T_1B_1B_2U_2$(short term)	2.3E-6	2.3E-6
Total	T_1B PT EST	2.59E-5	2.45E-6
	T_1B (long term)	2.35E-5 (91%)	ε (0%)
	T_1B (short term)	2.45E-6 (9%)	2.45E-6 (100%)

Notes:

1. In this example, removal or extension of the blackout-related time-dependent failure modes of RCIC and HPCS renders sequences 1, 2 and 4 successes and removes them from the calculated core damage frequency.

Table A.4 Comparison of the IDCOR and SNL Containment Matrices

Containment Failure Mode	Station Blackout		ATWS	
	IDCOR (T_1QUV)	SNL (TB)	IDCOR ($T_{23}C$)	SNL (TCSX)
No Containment Failure	.2	.08	--	.6
Pre-existing Leak or Failure to Isolate	.005	--	--	--
Overpressure Failure due Primarily to Hydrogen Burning	.6	.59	--	.2
Other Overpressure Failure	.2	.34	1.0	.2

<u>Drywell to Wetwell Leakage</u> (Station Blackout)	<u>SNL</u>	<u>IDCOR</u>
Design Leakage	.58	--
Late Penetration Failure	.34	--
Late Drywell Structural Failure	.02	--
Early Penetration Failure	.05	--
Early Drywell Structural Failure	.01	--

Table A.5 NRC/IDCOR Issues

Issue	Subject
1	Fission Product Release Prior to Vessel Failure
2	Recirculation of Coolant in Reactor Vessel
3	Release Model of Control Rod Materials
4	Fission Product and Aerosol Retention in the Primary System
5	In-vessel H_2 Generation
6	Core Slump, Core Collapse, and Reactor Vessel Failure
7	Containment Failure due to In-vessel Steam Explosions
8	Direct Heating of Containment
9	Ex-vessel Fission Product Release
10	Ex-vessel Heat Transfer Model from Molten Core to Concrete
11	Revaporization of Fission Products from the Primary System
12	Fission Product Deposition Model in Containment
13A	Suppression Pool Bypass (Pool Scrubbing)
13B	Retention of Fission Products in Ice Beds
14	Modeling of Emergency Response
15	Containment Performance
16	Secondary Containment Performance
17	Hydrogen Ignition and Burning
18	Essential Equipment Performance

Table A.6 Comparison of IDCOR and BCL Predictions of Fission Product Release for an ATWS Sequence With No Operator Actions Taken

Event	IDCOR*	NRC Contractors**
Containment Failure (HR)	1.0	1.3
Start of Core Melt (HR)	3.0	2.0
Vessel Failure (HR)	3.8	4.2
Fission Product Release Fractions***:		
Xe-Kr	1.0	1.0
I-Br	0.0008	0.003
Cs-Rb	0.0008	0.004
Te-Sb	0.0008	0.002
Sr	0.00001	0.002
Ba	0.00001	0.001
Ru-Mo	0.00001	Neg.
La	--	0.0001
Ce	--	0.0001

*IDCOR Technical Report 23.1GG, March 1985.
**NUREG/CR-4624, July 1986.
***Fraction of Initial Core Inventory.

Table A.7 Comparison of IDCOR and BCL Predictions of Fission Product Release for a Station Blackout Sequence With Hydrogen Burn

Event	IDCOR* (No Bypass)	NRC Contractors** (With Bypass)
Loss of Injection (HR)	0.0	6.0
Start of Core Melt (HR)	2.0	9.7
Vessel Failure (HR)	2.3	11.7
Containment Failure (HR)	47.0	11.7
Fission Product Release Fractions***:		
Xe-Kr	1.0	1.0
I-Br	7×10^{-5}	0.016
Cs-Rb	7×10^{-5}	0.013
Te-Sb	3×10^{-5}	0.11
Sr	1×10^{-5}	0.3
Ba	1×10^{-5}	0.18
Ru-Mo	1×10^{-5}	Neg.
La	--	0.021
Ce	--	0.034

*IDCOR Technical Report 23.1GG, March 1985.
**NUREG/CR-4624, July 1986.
***Fraction of Initial Core Inventory.

Table A.8 Comparison of IDCOR and SARRP Consequence Results (Person-Rem)

Accident Sequence	Containment Failure Mode	IDCOR	SARRP*
ATWS	Containment failure after core melt without significant pool bypass	$1.2x10^5$	~10^6
Station Blackout	Containment failure at RPV failure with significant pool bypass (TB2)	--	~10^7
Station Blackout	Early containment failure with RPV depressurization and drywell flooding (TBS)	--	~10^5
Station Blackout	Containment failure after a few hours without significant pool bypass	$1.2x10^5$	---
Station Blackout	Containment failure after a few hours without significant pool bypass (TB1)	--	~10^5
Station Blackout	Containment failure after many hours without significant pool bypass	$2.4x10^4$	---

*The SNL consequence calculations are not yet available and have been approximated based on release fractions in Tables A.6 and A.7.