725

【取扱い厳重注意】

平成２４年4月１６日

聴　取　結　果　書

東京電力福島原子力発電所における事故調査・検証委員会事務局
局　員　　三田　浩平

平成２４年４月１６日、 東京電力福島原子力発電所における事故調査・検証のため、関係者から聴取した結果は、下記のとおりである。

記

第１　被聴取者、聴取日時、聴取場所、聴取者等

１　被聴取者
原子力委員会委員長　近藤　駿介

２　聴取日時
平成２４年４月１６日午後１時３０分から同日午後３時１０分まで

３　聴取場所
霞が関合同庁舎４号館７階　原子力委員会委員長室

４　聴取者
事故調査委員会事務局　浅井主査、三田主査

５　ＩＣレコーダーによる録音の有無等
■　あり
□　なし

第２　聴取内容
１．不測事態シナリオ、日米協議について
２．PSA、外部事象のリスク、AM について
別紙のとおり

第３　特記事項
特になし

【取扱い厳重注意】

別紙

被聴取者に対して、事前に質問事項（別添１）を渡したところ、本聴取において、質問事項に対する回答を記した文書の提出があったので、同文書を添付する（別添２）。

１　不測事態シナリオ、日米協議について

同事項の回答については、別添２「１．不測事態シナリオに付いて」及び「２．日米協議について」のとおり。

以下に、補足説明内容を記す。

○　１．４）の補足

原子力委員会は、本来総理からの文書による諮問があれば、原子力委員会の合議を経て諮問に対して回答するのだが、総理は、口頭により不測事態シナリオ作成を私に依頼したため、私は、総理から専門家としての個人的見解を求められたと判断した。

また、原子力委員会は、原発事故への対処について所掌しておらず、原発事故への対処は、本来、原子力安全委員会の所掌である。総理からの不測事態シナリオ作成依頼は、原子力委員会の所掌範囲を越えているし、仮に総理からの依頼を私が原子力委員会として受けてしまったら、班目氏の顔を潰すことにもなる。

私は、不測事態シナリオの作成が原子力委員会の所掌範囲を越えていることと、総理が専門家としての個人的見解を私に求めたと判断したことから、私は、不測事態シナリオの作成依頼を原子力委員会としてではなく、私個人として受けた。

○　１．５）の補足

私は、細野補佐官に対して、不測事態シナリオの想定している事態が生ずる可能性はほぼないと思われるが、格納容器への窒素封入、高所からの注水装置に係る遠隔操作化、４号機使用済燃料プールの底部の強化等の整備を行えば、格段に安心度が増す旨説明した。

○　１．６）の補足

不測事態シナリオに記載した「本人が移転を希望するなら認めるべき区域」とは、避難を勧奨するべき地域ではなく、「本人が移転を希望するなら認めるべき区域」とは、チェルノブイリ事故の際に設定された避難の概念をそのままコピーしたものである。チェルノブイリ事故の際に設定された「本人が移転を希望するなら認めるべき区域」とは、当時のソ連市民に転居の自由が認められていたか否かは分からないが、確か、本人が希望をして移転したとしても、国がその移転に係る補償するというものであったと思う。

○　その他

（被聴取者に対して、燃料棒の損傷や溶融と水素発生の関係について、別添３の注釈１のとおりで間違いないかと質問したところ、）このとおりでよいと私は思う。

２　PSA、外部事象のリスク、AMについて

○　同事項の回答については、別添２ P.3「PSA、外部事象のリスク、AM について」のとおり。なお、別添２の P.6 の５）の根拠は別添４、別添２の P.6 の６）の根拠は別添５の P.8-16 右側３段目である。

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○ また、B.5.b については、民間事故調に私が説明した資料である、別添6の P.19 ～ 20 に記載のとおり、Dr. Nlils J. Diaz 氏が、平成23年10月の大阪の学会（ICONE）で講演していた。

以 上

【機密性２情報】

近藤原子力委員会委員長への質問事項

○　福島第一原発の不測事態シナリオの素描について

・　不測事態シナリオにおいて、避難範囲拡大の要因となる放射性物質放出は、主に４号機使用済燃料プールから放出されるものと理解してよろしいでしょうか。

・　近藤委員長が、（菅総理から不測事態シナリオ作成の依頼を受ける前の）平成２３年３月１５日から、事故の状況が更に悪化した場合の対応策について検討されていたのは何故でしょうか。

　→　３月１５日午前に４号機原子炉建屋爆発があり、４号機使用済燃料プールへの注水・使用済燃料の冷却に懸念が生じたからでしょうか。

・　平成２３年３月２２日、近藤委員長が菅総理から不測事態シナリオの作成を依頼された時、細野補佐官も、菅総理から何らかの指示を受けていたでしょうか。

・　不測事態シナリオ作成を原子力委員会としてではなく、個人として請け負ったのは何故でしょうか。

・　平成２３年３月２５日、近藤委員長が、菅総理ではなく細野補佐官に対して不測事態シナリオを提出されたのは何故でしょうか。

・　不測事態シナリオ P.15 の４ポツ、「移転を希望する場合認めるべき地域」とは、「移転を勧奨する地域」と理解してよろしいでしょうか。

○　日米協議について

・　平成２３年３月１８日午後の細野補佐官、長島衆議院議員、ルース駐日米国大使、NRC キャスト氏等の会談に参加されたでしょうか。

　→　参加されたとしたら、同会談でどのような話がなされたでしょうか。

・上記以外に、原発事故に関して、ルース大使と連絡をとられたでしょうか。

○　外的事象のリスク及び AM の認識、外的事象 PSA の技術水準等に関して

※　前回（平成２４年２月１日）のヒアリングにおけるご発言の趣旨を、数点確認させて頂きたいと思います。

「事故調ヒヤリングについて」にある質問への回答

1. 不測事態シナリオに付いて

1） そのとおりです。4号機のSFが溶融してコンクリートとの相互作用で多くの核分裂生成物を放出する場合、建家が損傷していて格納機能がないことから、１～３号機の炉心損傷時によりは多くの放射性物質が環境に放出されるとして解析しました。このデータの準備と解析はJAEA本間氏により行われました。

[redacted]

2） そうです。４号建屋の爆発と火災の報に接し、SF冷却悪化の可能性、再臨界等々について思いつく望ましくないシナリオのもたらす結果を当たっておくことにしたのです。取り敢えずは、何か都合の悪いことが起きるとすれば、どういう結果がどういうタイミングで生じるのかの当たりを付け、それまでに（そのときに）サイト内、サイト外でなにをしたらいいかを整理したいと、JAEAやJNESの専門家に課題を伝え、答えてくれるように依頼しました。

なお、この時間から少し後に、米国が８０km以内にいる米国人に避難命令を出したことについてNRCが計算結果をサイトにアップしたので、本間氏に彼らが何を想定してその判断に至ったのかを検討するよう依頼しました。国家安全保障会議のベーダー氏の近著にある、このときのホワイトハウスの対応の顛末を読むと、彼らもこの時点でほぼ同じモデルを使ったようです。

このような作業と事態の推移から、今後の不測事態として検討するべきシナリオが限定されてきたので、１９日になって、毎朝のように、細野補佐官、空本幹事、その他の国会議員諸氏が状況認識を共有するために私の部屋で一時間ほど行っていた自由討議の場に、わたしから、現場では今後こんなことに力をいれるべきという形でこの検討課題を話題にしました。

3） 総理は班目、寺坂、細野各氏と私の４人に「いろいろ役割はあるのだろうけれども協力してやってほしい」とした上で、「そろそろ落ち着いてきたので、最悪シナリオを考えてくれないか」と言われました。その際、このことについて受け答えをしたのは小生だけだったので、私が引き受けたという認識をもって、私と寺坂氏が退出しました。その間には総理と補佐官の間にやりとりはなかったと記憶しています。

4）　本件は原子力委員会が総理から諮問されたものとすれば、公開の席で審議・決定することになりますが、依頼の経緯からして専門家としての私に依頼されたものと理解しましたので、委員が行政部門や政務に対して専門家として個人的見解を述べるタイプの仕事に分類したのです。

5）　２２日に作業計画を細野補佐官に説明して作業を開始し、毎日進捗状況を伝え、２５日午後には中締めのメモをつくり、こんなところですと報告したところ、補佐官が「これでよい。このメモは一部のみとし、自分が預かって自ら総理に報告する」とされたので、作業は終了としました。

6）　地域の区別はチェルノビリ周辺で採用されたものを翻訳して用いています。英語で見るかぎり、「本人が移転を希望するなら認めるべき区域」となっていると思います。強制的に移転させるという概念と強制はしないが本人が希望するなら移転を認めるという概念が対になっているようです。家庭の事情で居住を続けるなら続けてもよいが、基本的には移転を勧奨するという積極さがあったかどうかはわかりません。

2.　日米協議について

1）　3月18日は、ルース大使に対して細野補佐官が日本政府としてこういう専門家の助言を得てきちんと対応していると説明し、私と久木田さんがそれぞれ状況認識を述べて終わりました。短い時間の会合で、キャスト氏は発言しなかったと記憶しています。

2）　本件に関して最初にルース大使とコンタクトを持ったのは、日曜日朝と思います。オフィスに電話をもらい、ポネマンDOE副長官が話したがっているということで、大使館の交換機経由で彼と会話、彼に小生の状況認識を伝えました。その際、NRCのヤッコ委員長が話したがっているので電話してくれといわれたのですが、彼のカウンターパートは班目委員長だから、安全委員会に電話してくれと伝えるようにと返しました。

3）　その後、念のため、安全委員会事務局（委員は不在だった）にそのことを伝えると、NRCから電話がすでにあったとのことだったので、適切に対応するよう私が言っていたと委員長に伝えるように依頼しました。私としてはDOEとNRCの両方のコンタクトを維持することは無理だし、当然安全委員会が対応するべきものと考えたからです。この事情は後でヤッコ委員長に説明しました。

4） IAEAでINES制度を作るのに協力したものとして、こうしたときに海外への通報責任はINESナショナルオフィサーが担うものと思っていました（そのことが災害対策本部の機能として認識されていたか、チェックしていませんが）。他方、海外ではこういうときにトップが直接カウンターパートに電話することが多いことの認識も制度設計において考慮することも重要と感じています。

5） ポネマン氏とはその後電話ではもう一度（多分翌日）やり取りしたが、その際、米側からの支援可能リストを電話口で言われたのには降参。DCの大使館にそのリストを入れてくれと頼み、またDOEのカウンタパートであるエネ庁に提案があったことを伝えることを約束しました。その後、エネ庁長官に電話したところ、長官は既にいろいろ聞いている、趣旨はわかったということでした。後刻、私としては、本当に欲しいものは東電が選択するべきと感じ、保安院根井審議官に対して東電にその旨伝えることをメールで依頼しました。根井氏からは大使館ルートでそのような話が東電に伝えているが、東電の応答は鈍いという趣旨の返事がきました。

6） その後は29日早朝、来日したヤッコ委員長と懇談した際に大使が同席していましたが、これは儀礼的な挨拶だけでした。私から委員長に対して、既に日米協議がはじまっていたことを踏まえてお礼をいい、これに対して先方からは困ったことがあったら何でも直接自分に行ってくれればなんとかするといわれました。しかし、このNRCルートは既に政府ベースで活動していましたから、私はその後関与していません。私はDOEルートを東電を入れた専門家の対話のチャネルにするべく、電話会議を志向し、実現しました（米国側にはチューDOE長官を始め、いろいろな組織の人が参加、聴いていたようです）。

PSA、外部事象のリスク、AMについて

1） 我が国では、「PSAは不完全であり、不確実性が大きく未熟で規制に使えない。規制判断には保守的な評価を行う確定論的アプローチを用いるべし」との言説が支配的であった。保守的といっても確率論的に評価しない限り主観的。それは専門家が保守性の判断を独占する不透明なアプローチと返してきたが、安全委員会等においては、長く、かならずそういった修飾語を冠して扱われてきた。

他方、非原子力界からは、確率的安全目標を定め、これを使って対策の十分性を判断する取組に対して、災害ポテンシャルが大きいものは最悪に備えるべきであり、確率が小さいからといって最悪シナリオを切り捨てるのは間違いと批判されてきた。これには、「そもそも最悪シナリオというけれど、それなりに蓋然性を推定している。だから、あなたが最悪シナリオを提示してくれたら、私はそれよりもっと悪いシナリオを必ず思いついてあげますよ。How safe is safe enough? に対する取組は、できるだけ多くの失敗なり、異常現象の情報を集め、望ましくない結果をもたらすシナリオを人智を尽くして列挙し、目標を満たさないシナリオに対策を施し、修正されたシステムについて再びシナリオを尽くし、目標を満たさないシナリオに対策を施すことを繰り返していくのです。この作業をどこで打ち切るか、その判断基準が安全目標なのですとしてきた。そうすると「それはわかったけれど、放射線被ばくはとにかくいや。大事故の発生確率は巨大隕石の落下で東京がなくなる確率ぐらいに低くないといや」という人が出てきたこともあります。専門家は、それが一年のうちに発生するチャンスは一億分の一というから、それでは貴方の提案はそういう水準に安全目標をおくことですねといいつつ、議論を続けることになります。これを目標にするとこの装置では万に一つのチャンスでこんなことがあり得るとわかった場合に、そのまれな事象が起きた際にもこの装置があれば、万にひとつも被害の発生には至らないといえる、そんな装置を設置することが必要ということになりますから、そんなシナリオを何百と当たって、合計する訳ですから、総合的には、百万分の一ぐらいが人智の限界かなと言う気がしていますが。

2） こうした作業における私の最大の誤りは、安全目標の議論において公衆の過剰被ばくの発生確率を指標に選んだこと。命を守ることが安全だと思い込んでいた。公衆の過剰被ばくのリスクはIAEAの安全目標と整合するし、交通事故等による死亡リスクを参照しつつ、それより十分小さいことをもって合意を求める議論を展開しやすい。しかも、敷地境界の公衆の個人の線量は、レベル３PSAを行わないでも、レベル２PSAで評価ができてしまうことから、安全目標の制定論議をいそぐにも便利だったので、これを選んだが、レベル３PSAに基づく敷地境界をこえた土地汚染の発生確率を選ぶべきであった。これによるコミュニティ崩壊の深刻さ、さらにはオフサイトセンターの設置場所に関する教訓をチェルノビリ事故から学ばなかったことは本当に不覚であった。

遅くとも、JCO事故のときに１０km圏内に屋内退避まがいの勧告を発したり、乾し芋その他の産品に関する風評被害が発生したのを見て、防災対策まで

考慮して死亡リスクが小さいから合格という安全確保体系は社会の受け入れるところではないと悟るべきであった。中央防災会議は津波対策の議論で強固な堤防建設による防災と土地利用の制限による減災を比較して後者を選んだが、原子力発電所については事故を想定しての敷地外の土地利用の制限は選べない、よってその必要性が生じる確率が百万分の一もないように、原子炉にFCVSの設置を求めることを選ぶべきであった。

JCO 事故の後の原子力防災の議論においては、制度設計に時間を使い、官邸の役割を強化したのは間違いではないと思っているが、レベル3PSA を使ってJCO事故のトレースを行い、防災対策の効能をチェックし、この経験を踏まえて、我ら何を目指すべきかを議論するべきだった。ラコステ氏にお前達はJCO事故から何も学ばなかったのではと言われたが、あのときは、核燃料サイクル施設の規制お粗末という反応で忙しく、事故がおきて人々が反応した貴重な事例であるとして、安全目標の指標の在り方を考えるとか、防災計画の在り方の見直しに思いが至らなかった。

3) 北欧においてはチェルノビリ事故のあと、直ちに格納容器過圧破損防止のために FCVS が設置されたが、これは現実に土地汚染を経験した社会において原子力発電が生き残るためには、土地汚染の可能性を排除する決意表明が必要との認識があったからと理解していた。

当時、FCVS については、そもそも期待通り機能するのか、水素燃焼対策に疑問がある技術であること、過圧破損を防止するには、これよりは格納容器内における冷却機能を強化してそもそも過圧事象を発生しないようにする方が総合的に考えて合理性があるのではという議論、さらには、事故時の大規模な放射性物質の放出シナリオには格納容器バイパス事象もあるところ、FCVS は希ガスをフィルターできないから、バイパスラインの追加にも見える。よって、土地汚染の防止というが、被ばくリスク低減の観点からの役割は限られ、設置の合理性には疑問があるという議論がなされた。これに対して、設置者は、頻度は低くても、制御できない土地汚染物質の放出シナリオをなくすことが原子力発電の生き残りの必要条件と思い定めて付けているということであった。

こうした議論を経るに、我々は、彼らはこの設置を政治的に選択したと受け止めた。他方、米国や英国では、上の論理でこれの設置に合理性を見いだせない、わずかな可能性である土地汚染については防災対策でカバーするという立場であった。実際、米国では、いまでも PWR については FCVS は設置されていないし、福島事故後の対策にも予定はない。

我々は、このように土地汚染にたいする嫌悪感の違いを目撃していたのだが、これを我が身の問題として自省することなく、安全目標の考え方を決める際に米国の考え方を採用し、JCO 事故を経験しても、これを見直さなかったのは不覚と言わざるを得ない。

4) 地震 PSA の結果で当時参照可能だったのは NUREG１１５０。この結果から、地震リスクは内部事象リスクと同様かそれを超えることもあること、而して、地震 PSA 実施の最大の課題は地震学者に地震ハザード曲線の作成に協力頂くことであることがわかったので、当時の NUPEC/安全解析所においてその実施努力を行っていただいた。

この地震ハザード曲線作業の難しさは、断層の性質等に異なる見解がある時には、それぞれの見解に学界の意見で重み付けして全て取り入れるというアプローチにつきあうことに対する嫌悪感を地震学者に克服していただくことであった。真実はひとつという学者の立場と相容れないと忌避されることが多く、決定しなければならない時には、学説を戦わせた後は、こうした方法によってそれぞれの意見をそれなりに生かすことには、それなりに意味があるとしてつきあっていいよと言っていただけるまでが大変であった(今も津波ハザード曲線作りにおいて関係学者の協力を得るのに苦労しているらしい)。

5) NUREG１１５０の１３−５ページには、次ページに示すいくつかの AM の感度分析の結果がある。これは地震 PSA を踏まえたものではないが、それぞれの手段が炉心損傷確率を半減する程度の意味を持っていることが示唆されている。これが我々が、しばしば「AM は既に小さい事故確率をさらに小さくするための工夫であって、総合的に検討し、整備されるべきもの」としてきた根拠である。

6) 我が国では、私が関係していた時代には、地震 PSA に基づく AM の効果の評価は勉強した記憶はない。AM 用機器設備の耐震クラスをどうするかは議論され、実力として高いクラスのものであるべきとしたはず。NUREG１１５０にはSSE の４倍くらいで電源等が故障するとあるが、このクリップフエッジの内側では、地震時であろうとも内部事象 PSA による AM 評価の結果は使えると思料。耐震性は国内でもそんなところと思うが、このあたりはストレステストの結果まち。

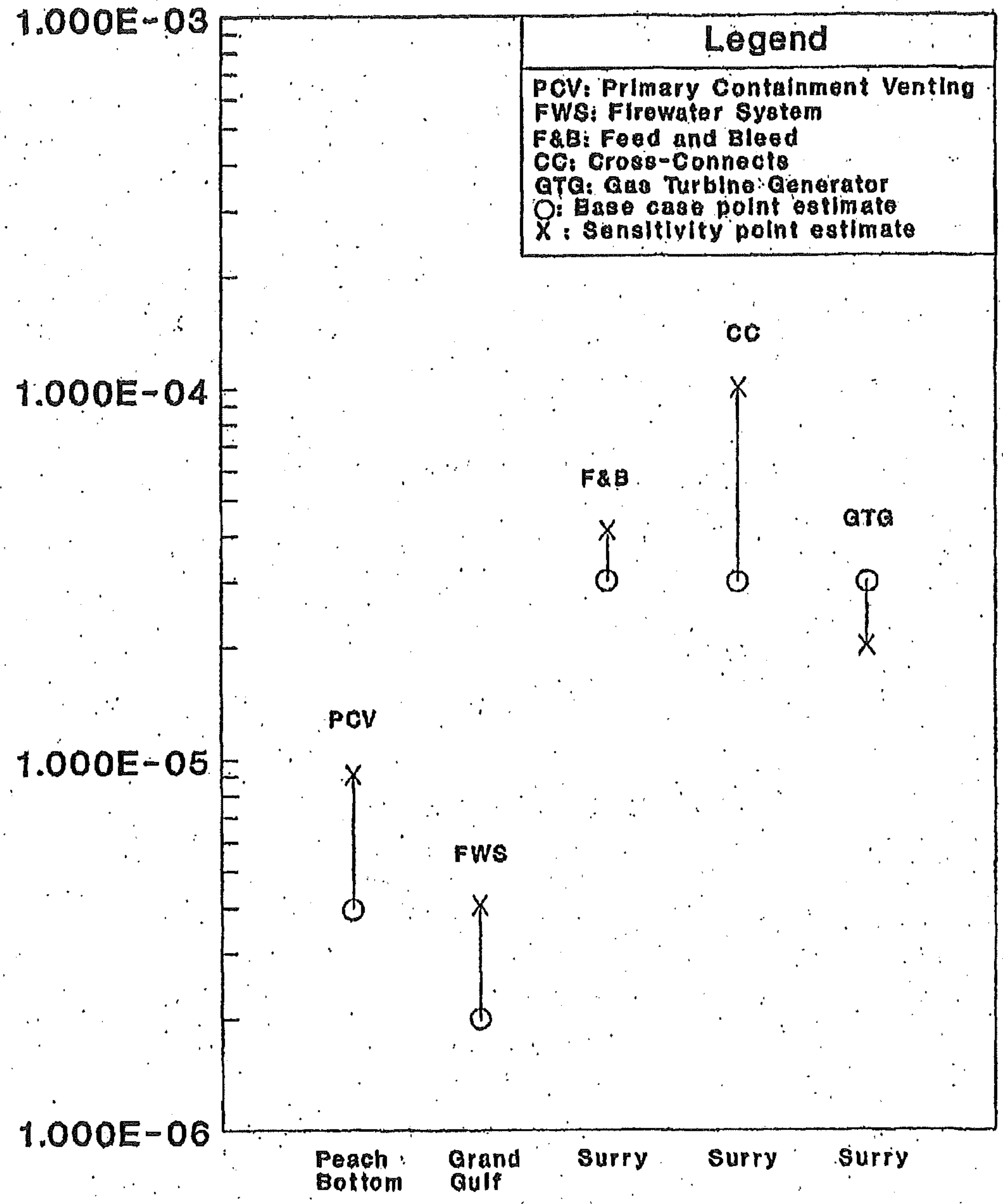
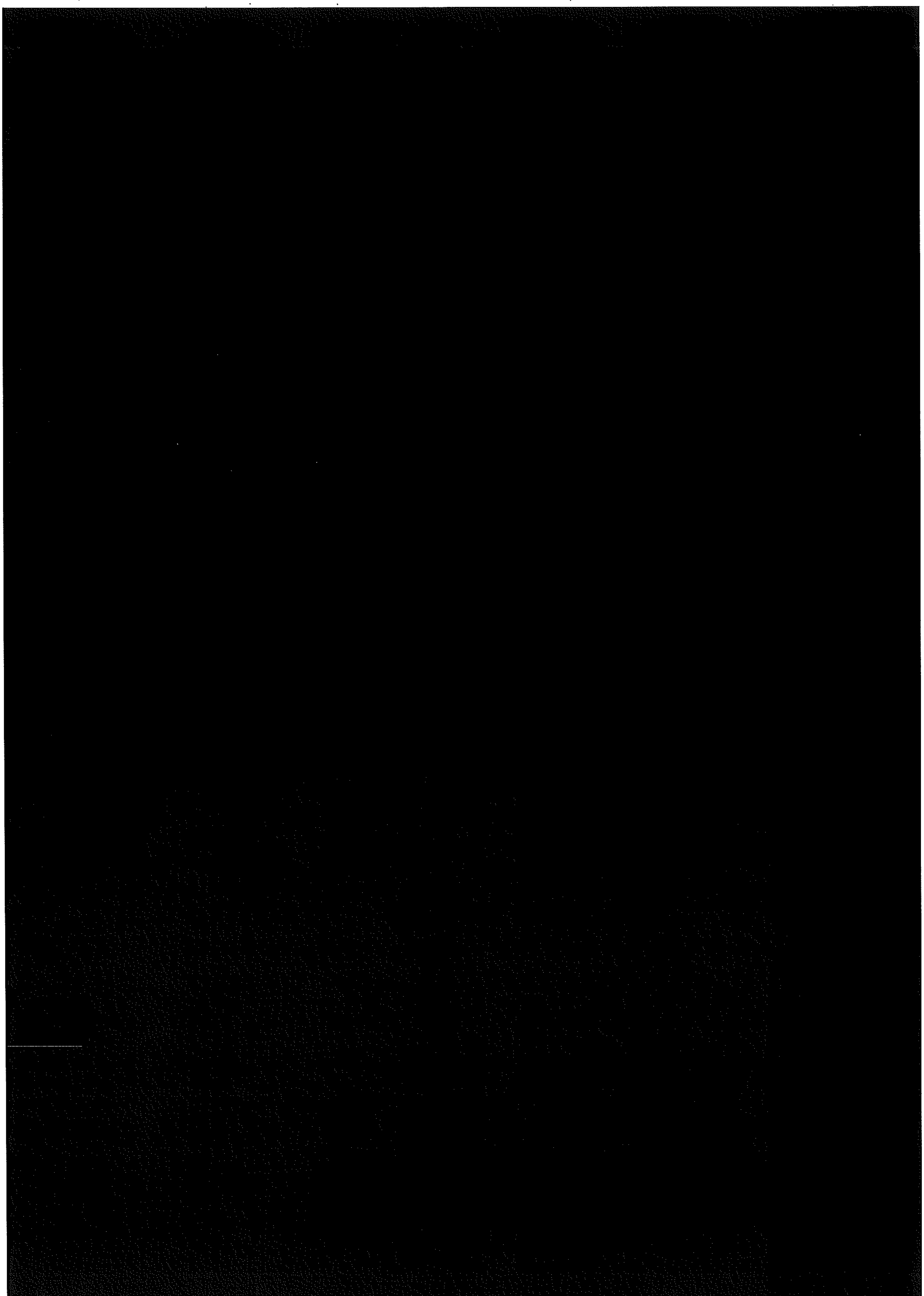
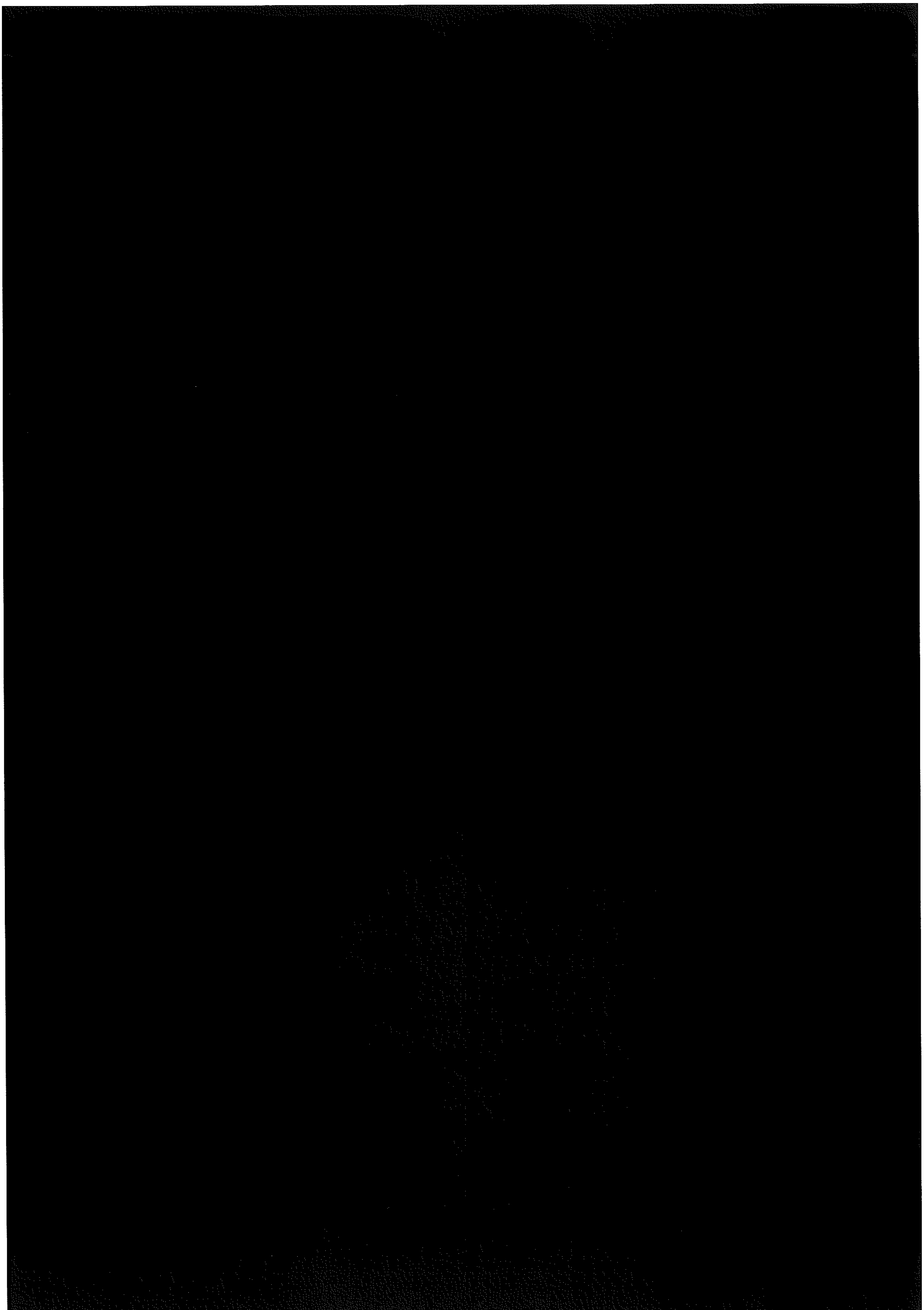


Figure 13.1 Benefits of accident management strategies.





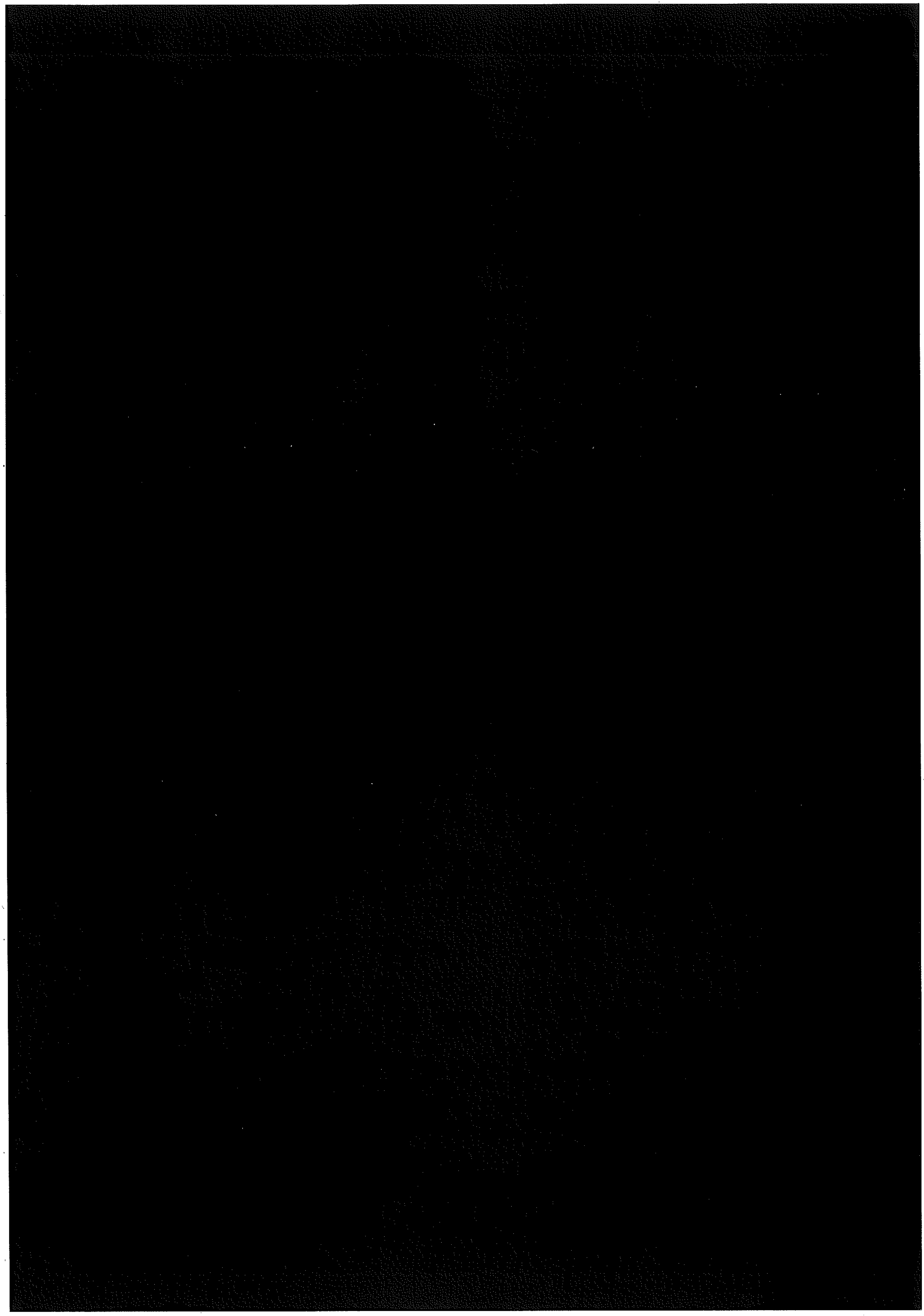


Table 13.1 Utility of NUREG-1150 PRA process to other plant studies.

Example Results	Applicability	
	Class of Plants	Plant Population
1. Methods (e.g., uncertainty, elicitation, event tree/fault tree)	high	high
2. General perspectives (e.g., principal contributors to core damage frequency and risk)	medium	low
3. Supporting data base on design features, operational characteristics, and phenomenology (e.g., hydrogen generation in core damage accidents, operational data)	high	medium
4. Quantitative results (e.g., core damage frequency, containment performance, risk)	low	low

Site-specific requirements and differing utility requirements often lead to significant differences in support system designs (e.g., ac power, dc power, service water) that can significantly influence the response of the plant to various potential accident-initiating events. Further, different operational practices, including maintenance activities and techniques for monitoring the operational reliability of components or systems can have a significant influence on the likelihood or severity of an accident.

13.2.2 Guidance for Accident Management Strategies

Certain preparatory and recovery measures can be taken by the plant operating and technical staff that could prevent or significantly mitigate the consequences of a severe accident. Broadly defined, such "accident management" includes the measures taken by the plant staff to (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) maintain containment integrity as long as possible, and finally (4) minimize the consequences of offsite releases. In addition, accident management includes certain measures taken before the occurrence of an event (e.g., improved training for severe accidents, hardware or procedure modifications) to facilitate implementation of accident management strategies. With all these factors taken together, accident management is viewed as an important means of achieving and maintaining a low risk from severe accidents.

Under the staff program, accident management programs will be developed and implemented by licensees. The NRC will focus on developing the regulatory framework under which the industry programs will be developed and implemented, as well as providing an independent assessment of licensee-proposed accident management capabilities and strategies. NUREG-1150 has been used by the NRC staff to support the development of the accident management program. NUREG-1150 methods provide a methodological framework that can be used to evaluate particular strategies, and the current results provide some insights into the efficacy of strategies in place or that might be considered at the NUREG-1150 plants. Thus, the NUREG-1150 methods and results will support a staff review of licensee accident management submittals.

PRA information has been used in the past to influence accident management strategies; however, the methods used in NUREG-1150 can bring added depth and breadth to the process, along with a detailed, explicit treatment of uncertainties. The integrated nature of the methods is particularly important, since actions taken during early parts of an accident can affect later accident progression and offsite consequences. For example, an accident management strategy at a BWR may involve opening a containment vent. This action can affect such things as the system response and core damage frequency, the retention of radioactive material within the containment, and the timing of radionuclide releases (which impacts evacuation strategies). It is possible that actions to reduce the core damage frequency can yield accident sequences of lower frequency but with much higher consequences. All these factors need to be considered in concert when developing appropriate venting strategies. The treatment of uncertainties is another key aspect of accident management. Generally, procedures are developed based on "most likely" or "expected" outcomes. For severe accidents, the outcomes are particularly uncertain. PRA models and results, such as those produced in the accident progression event trees, can identify possible alternative outcomes for important accident sequences. By making this information available to operators and response teams, unexpected events can be recognized when they occur, and a more flexible approach to severe accidents can be developed. The recent trend toward symptom-based, as opposed to event-based, procedures is consistent with this need for flexibility.

To demonstrate the potential benefits of an accident management program, some example calculations were performed, as documented in Reference 13.20. For this initial demonstration, these calculations were limited to the internal-event accident sequence portion of the analysis. Further, the numerical results presented are "point estimates" of the core damage frequency as opposed to mean frequency estimates. Selected examples from the initial analysis are presented below.

Effect of Firewater System at Grand Gulf

The first NUREG-1150 analysis of the Grand Gulf plant (Ref. 13.21) did not credit use of the firewater system for emergency coolant injection because of the unavailability of operating procedures for its use in this mode and the difficulties in physically configuring its operation. However, since that time, the licensee has made significant system and procedural modifications. As a result, the firewater system at Grand Gulf can now be used as a backup source of low-pressure coolant injection to the reactor vessel. The system would be used for long-term accident sequences, i.e., where makeup water was provided by other injection systems for several hours before their subsequent failure. The firewater system primarily aids the plant during station blackout conditions and is considered a last resort effort.

An examination has been made of the benefit of these licensee modifications to the Grand Gulf plant. As shown in Figure 13.1, these analyses showed that the total core damage frequency was reduced from 4E-6 to 2E-6 per reactor year because of these changes.

Effect of Feed and Bleed on Core Damage Frequency at Surry

The NUREG-1150 analysis for Surry includes the use of feed and bleed cooling for those sequences in which all feedwater to the steam generators is lost (thus causing their loss as heat removal systems). Feed and bleed cooling restores heat removal from the core using high-pressure injection (HPI) to inject into the reactor vessel and the power-operated relief valves (PORVs) on the pressurizer to release steam and regulate reactor coolant system pressure.

An examination has been made to determine to what extent feed and bleed cooling decreases core damage frequency at Surry. The current Surry model includes two basic events representing failure modes for feed and bleed cooling in the event of a loss of all feedwater. These modes are: operator failure to initiate high-pressure injection and operator failure to properly operate the PORVs. In order to examine the impact of feed and bleed cooling, both basic events were assumed to always occur. As shown in Figure 13.1, the resulting total core damage frequency for Surry (if feed and bleed cooling were not available) then increases by roughly a factor of 1.3. That is, the availability of the feed and bleed core cooling option in the Surry design and operation is estimated to reduce core damage frequency from 4E-5 to 3E-5 per reactor year.

Gas Turbine Generator Recovery Action at Surry

The present NUREG-1150 modeling and analysis of the Surry plant have not considered the benefits of using onsite gas turbine generators for recovery in the event of station blackout accidents. Both a 25 MW and a 16 MW gas turbine generator are available to provide emergency ac power to safety-related and non-safety-related equipment. These generators were not included in the analysis because, as currently configured, they would not be available to mitigate important accident sequences.

An examination has been made of the effect on core damage frequency at Surry of including the gas turbine generators as a means of recovery from station blackout sequences. To give credit for the addition of one generator for emergency ac power, it is assumed that Surry plant personnel have the authority to start the gas turbines when required and that 1 hour is required to start the gas turbines and energize the safety buses. In the analysis, the gas turbines were assumed to be available 90 percent of the time.

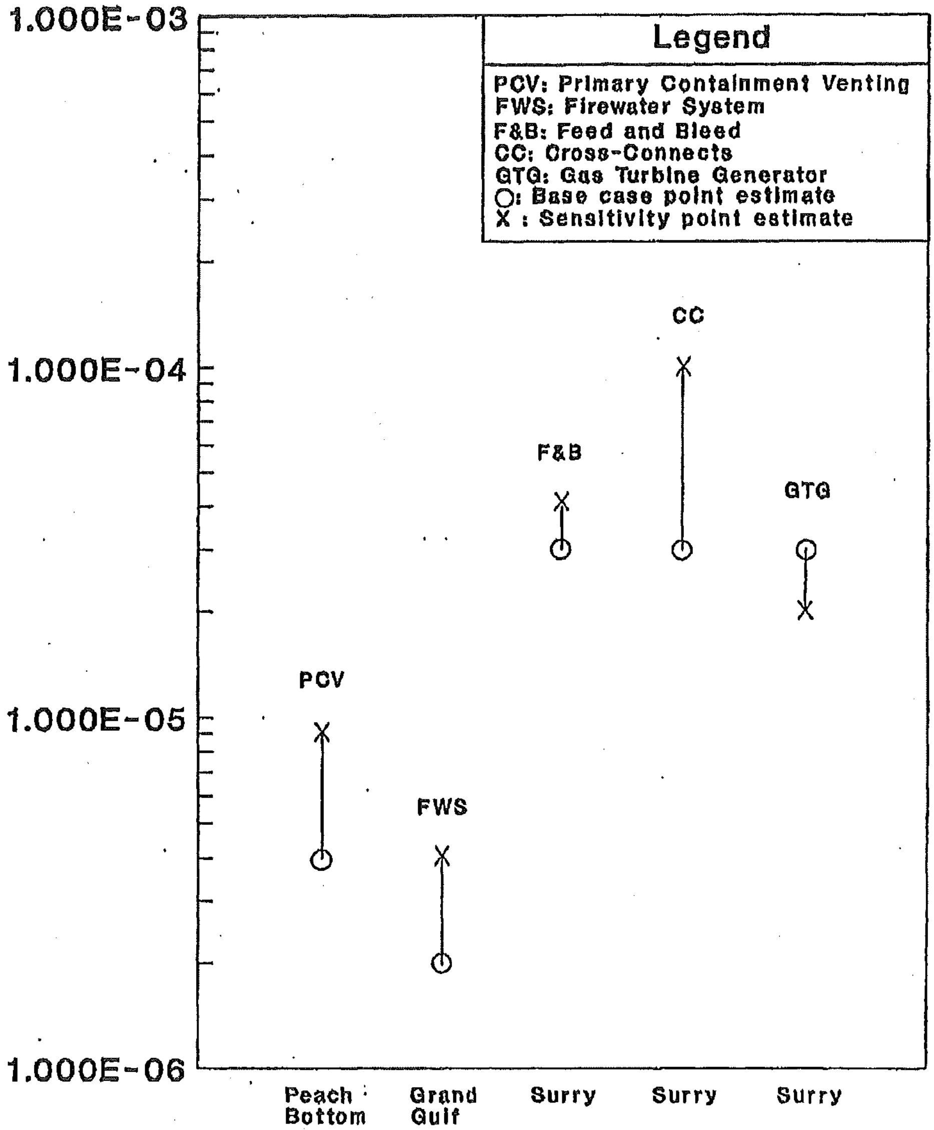


Figure 13.1 Benefits of accident management strategies.

The use of the onsite gas turbine was estimated to reduce core damage frequency from 3E-5 to 2E-5 per reactor year.

High-Pressure Injection and Auxiliary Feedwater Crossconnects at Surry

The Surry Unit 1 plant is configured to recover from loss of either the high-pressure injection (HPI) system or the auxiliary feedwater (AFW) system by operator-initiated crossconnection to the analogous system at Unit 2. While these actions provide added redundancy to these systems, new failure modes (e.g., flow diversion pathways) that were included in the modeling process for Surry have been created. The alignment of the Unit 1 and Unit 2 HPI and AFW systems for crossconnect injection is modeled as a recovery action.

Analysis of the importance of crossconnect injection at Surry includes two parts. First, credit for crossconnect injection was removed from all applicable dominant sequences, which were then requantified. Second, sequences that were previously screened out of the analysis were checked to determine if they would become dominant in the absence of crossconnect injection. As shown in Figure 13.1, the point estimate of the total core damage frequency without crossconnects is 1E-4, compared to the value of 3E-5 for internally initiated events in the base case.

Primary Containment Venting at Peach Bottom

The primary containment venting (PCV) system at Peach Bottom is used to prevent primary containment overpressurization during accident sequences in which all containment heat removal is lost. Most sequences of this type involve failure of the residual heat removal systems. Because of the existence of this venting capability, no such accident sequences appeared as dominant in the NUREG-1150 analysis for Peach Bottom.

The effect of the PCV system on the core damage frequency at Peach Bottom was determined by examining the sequences screened out in the NUREG-1150 analysis that included the PCV system as an event (primarily the sequences involving loss of containment heat removal). Credit for the PCV system was removed from these sequences, which were then summed and added to the current point estimate of the core damage frequency. As shown in Figure 13.1, this results in a point estimate of the Peach Bottom core damage frequency without containment venting of 9E-6, about a factor of 2.6 increase over the NUREG-1150 value of 4E-6.

13.2.3 Improving Containment Performance

The NRC has performed an assessment of the need to improve the capabilities of containment structures to withstand severe accidents (Ref. 13.1). Staff efforts focused initially on BWR plants with a Mark I containment, followed by the review of other containment types. This program was intended to examine potential enhanced plant and containment capabilities and procedures with regard to severe accident mitigation. NUREG-1150 provided information that served to focus attention on areas where potential containment performance improvements might be realized. NUREG-1150 as well as other recent risk studies indicate that BWR Mark I risk is dominated by station blackout and anticipated transient without scram (ATWS) accident sequences. NUREG-1150 further provided a model for and showed the benefit of a hardened vent for Peach Bottom (discussed above and displayed in Figure 13.1). The staff is currently pursuing regulatory actions to require hardened vents in all Mark I plants, using NUREG-1150 and other PRAs in the cost-benefit analysis.

The NUREG-1150 accident progression analysis models were used by the staff and its contractors in the evaluation of possible containment improvements for the PWR ice condenser and BWR Mark III designs. The result of the staff reviews of these designs (and all others except the Mark I) was that potential improvements would best be pursued as part of the individual plant examination process (discussed in Section 13.2.1).

13.2.4 Determining Important Plant Operational Features

NUREG-1150 will provide a source of information for investigating the importance of operational safety issues that may arise during day-to-day plant operations. The NUREG-1150 models, methods, and results have already been used to analyze the importance of venting of the suppression pool, the importance of keeping the PORVs and atmospheric dump valves unblocked, the importance of operational characteristics of the ice condenser containment design, the importance of operator recovery during an accident sequence, and the importance of crossties between systems. These operational and system characteristics, as well as many others, are described in detail in Chapters 3 through 7. For example, characteristics of the Surry plant design and operation that

8. PERSPECTIVES ON FREQUENCY OF CORE DAMAGE

8.1 Introduction

Chapters 3 through 7 have summarized the core damage frequencies individually for the five plants assessed in this study. Significant differences among the plants can be seen in the results, both in terms of the core damage frequencies and the particular events that contribute most to those frequencies. These differences are due to plant-specific differences in the plant designs and operational practices. Despite the plant-specific nature of the study, it is possible to obtain important perspectives that may have implications for a larger number of plants and also to describe the types of plant-specific features that are likely to be important at other plants. This chapter provides some of these perspectives.

8.2 Summary of Results

As discussed in Chapter 2, the core damage frequency is not a value that can be calculated with absolute certainty and thus is best characterized by a probability distribution. It is therefore discussed in this report in terms of the mean, median, and various percentile values. The internal-event core damage frequencies are illustrated graphically in Figure 8.1 (Refs. 8.1 through 8.5). The figure does not include the contributions of external events, which are discussed in Section 8.4.

In Figure 8.1 the lower and upper extremities of the bars represent the 5th and 95th percentiles of the distributions, with the mean and median of each distribution also shown. Thus, the bars include the central 90 percent of the distributions (it should be remembered that the distributions are not uniform within these bars). These figures show that the range between the 5th and 95th percentiles covers from one to two orders of magnitude for the five plants. There is also significant overlap among the distributions, as discussed below. The reader should refer to References 8.1 through 8.5 for detailed discussion of the distributions.

Figures 8.2 and 8.3 show the contributions of the principal types of accidents to the mean core damage frequency for each plant. Figure 8.4 also presents this breakdown, but on a relative scale. These figures show that some types of accidents, such as station blackouts, contribute to the core damage frequencies for all the plants; however, there is substantial plant-to-plant variability among important accident sequences.

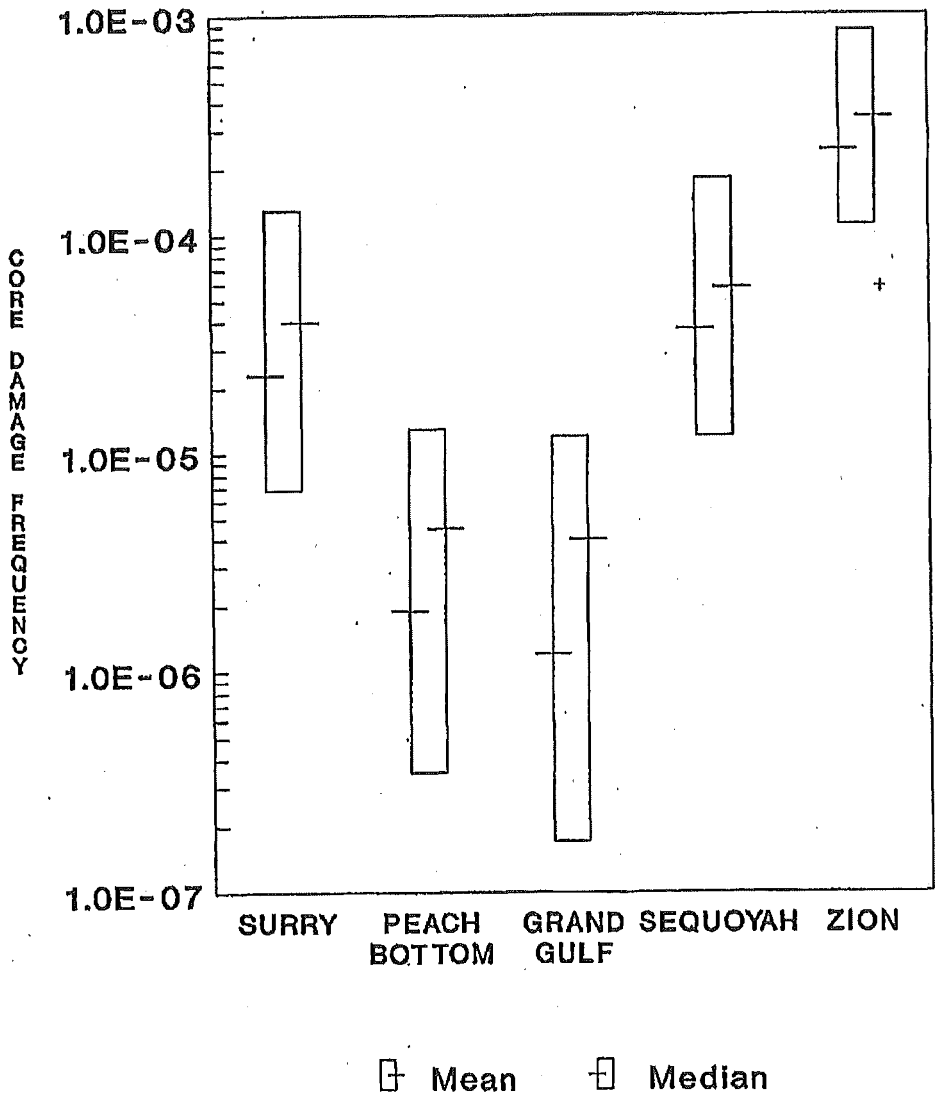
Figures 8.5 through 8.8 provide the results of the external-event analyses, and Figures 8.9 through 8.12 give the breakdown of these analyses according to the principal types of accident sequences.

8.3 Comparison with Reactor Safety Study

Figures 8.13 and 8.14 show the internal core damage frequency distributions calculated in this present study for Surry and Peach Bottom along with distributions synthesized from the Reactor Safety Study (Ref. 8.6), which also analyzed Surry and Peach Bottom. The Reactor Safety Study presented results in terms of medians but not means. It can be seen that the medians are lower in the present work, although observation of the overlap of the ranges shows that the change is more significant for Peach Bottom than for Surry.

There are two important reasons for the differences between the new figures and those of the Reactor Safety Study. The first is the fact that probabilistic risk analyses (PRAs) are snapshots in time. In these cases, the snapshots are taken about 15 years apart. Both plants have implemented hardware modifications and procedural improvements with the stated purpose of increasing safety, which drives core damage frequencies downward.

The second reason is that the state of the art in applying probabilistic analysis in nuclear power plant applications has advanced significantly since the Reactor Safety Study was performed. Computational techniques are now more sophisticated, computing power has increased enormously, and consequently the level of detail in modeling has increased. In some cases, these new methods have reduced or eliminated previous analytical conservatisms. However, new types of failures have also been discovered. For example, the years of experience with probabilistic analyses and plant operation have uncovered the reactor coolant pump seal failure scenario as well as intersystem dependencies, common-mode failure mechanisms, and other items that were less well recognized at the time of the Reactor Safety Study. Of course, this same experience has also uncovered new ways in which recovery can be achieved during the course of a possible core damage scenario (except for the



Notes: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

"+" indicates recalculated Zion mean core damage frequency based on recent plant modifications (see Section 7.2.1).

Figure 8.1 Internal core damage frequency ranges (5th to 95th percentiles).

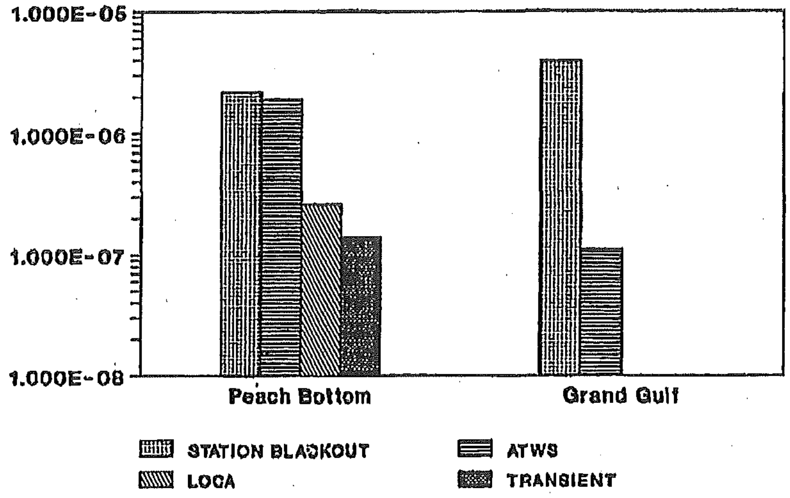
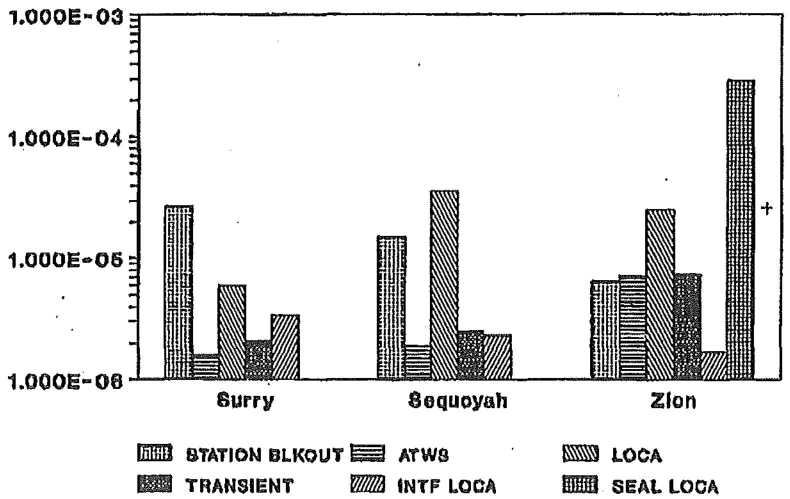


Figure 8.2 BWR principal contributors to internal core damage frequencies.



Notes: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

"+" indicates recalculated mean seal LOCA plant damage state frequency based on recent plant modifications (see Section 7.2.1).

Figure 8.3 PWR principal contributors to internal core damage frequencies.

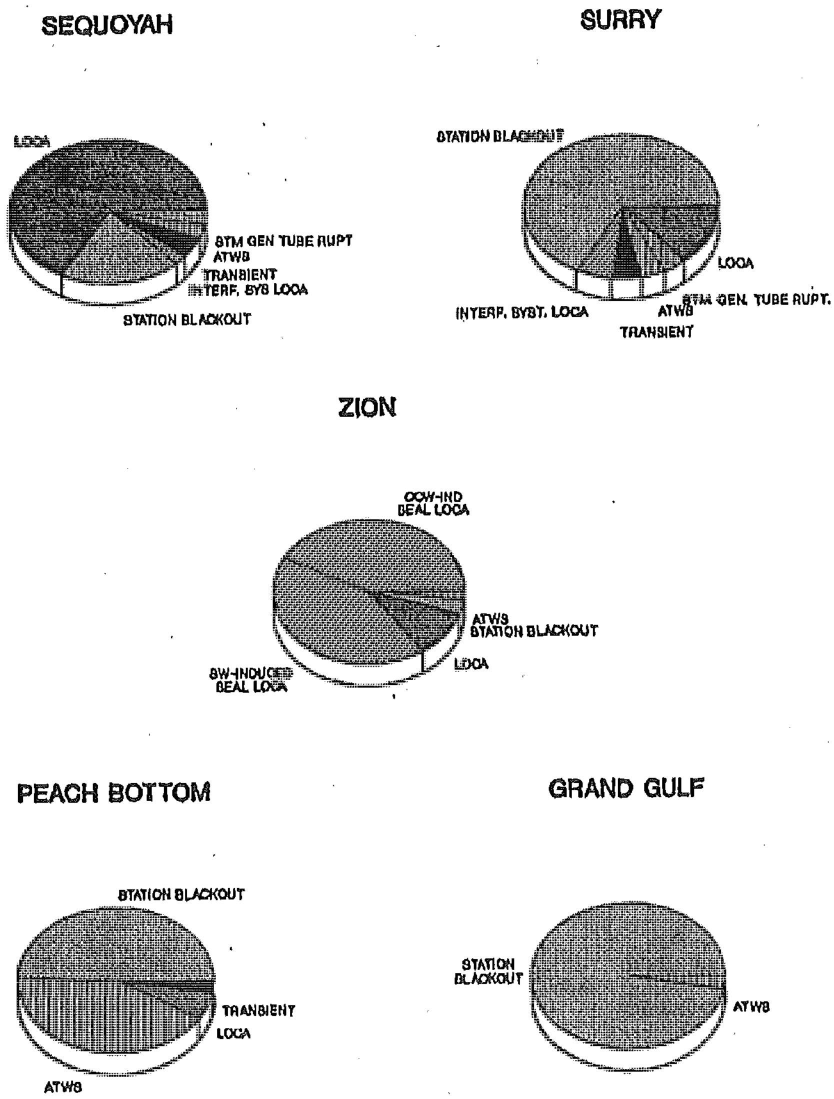


Figure 8.4 Principal contributors to internal core damage frequencies.

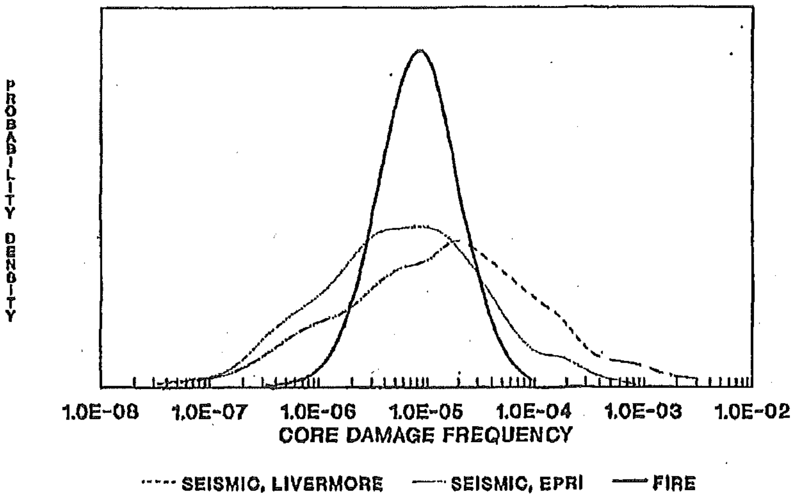
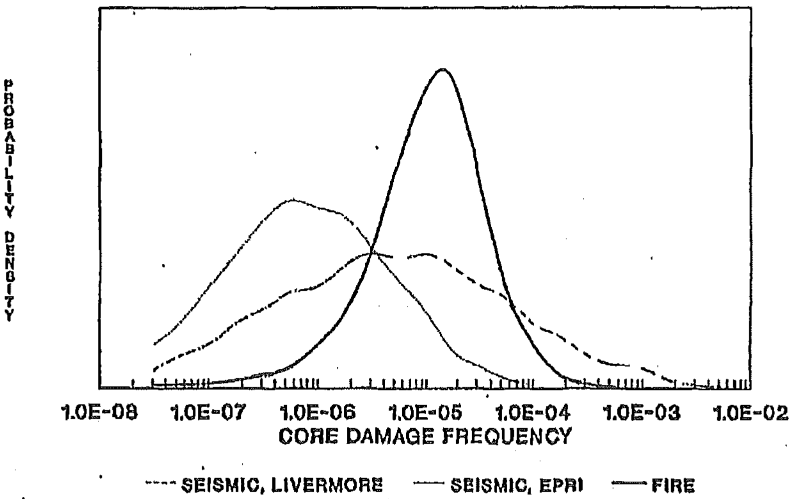


Figure 8.5 Surry external-event core damage frequency distributions.



Note: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Figure 8.6 Peach Bottom external-event core damage frequency distributions.

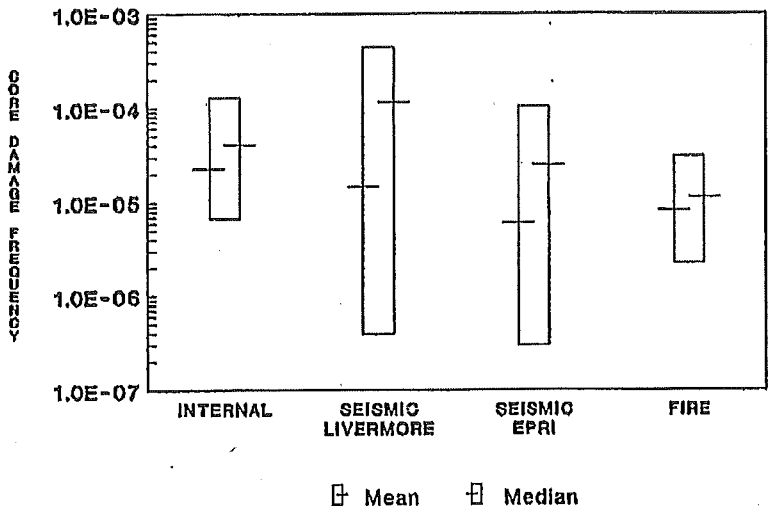
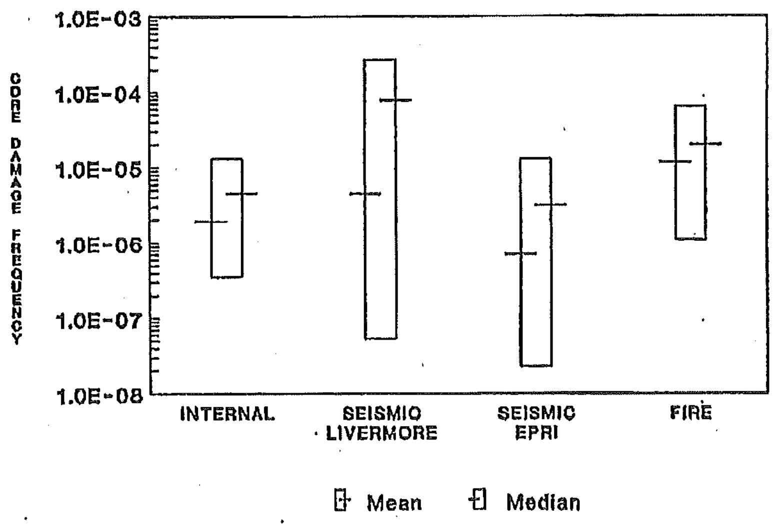


Figure 8.7 Surry internal- and external-event core damage frequency ranges.



Note: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Figure 8.8 Peach Bottom internal- and external-event core damage frequency ranges.

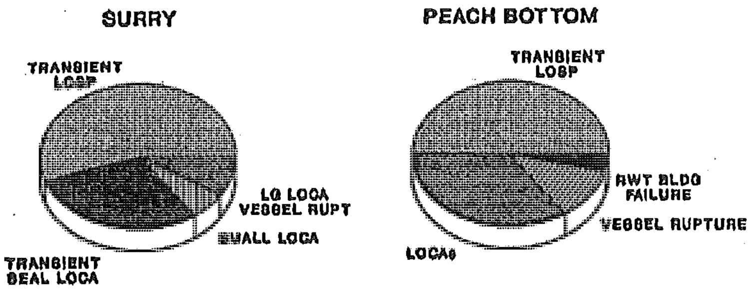


Figure 8.9 Principal contributors to seismic core damage frequencies.

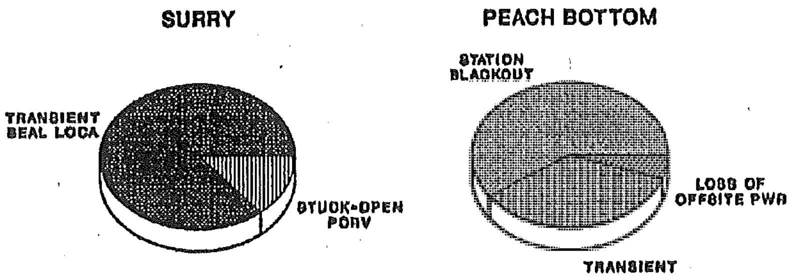


Figure 8.10 Principal contributors to fire core damage frequencies.

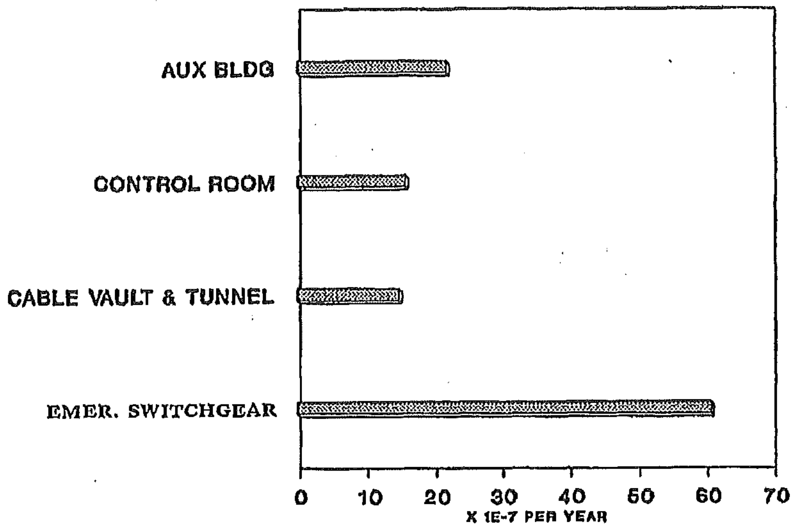
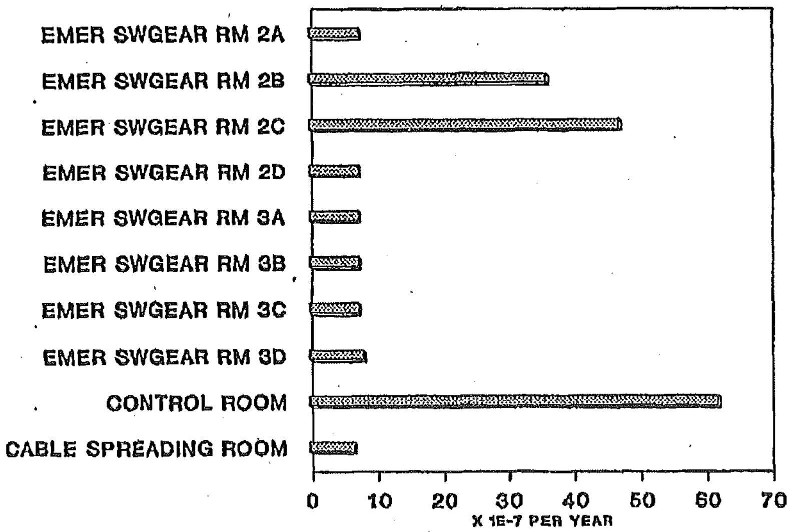


Figure 8.11 Surry mean fire core damage frequency by fire area.



Note: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Figure 8.12 Peach Bottom mean fire core damage frequency by fire area.

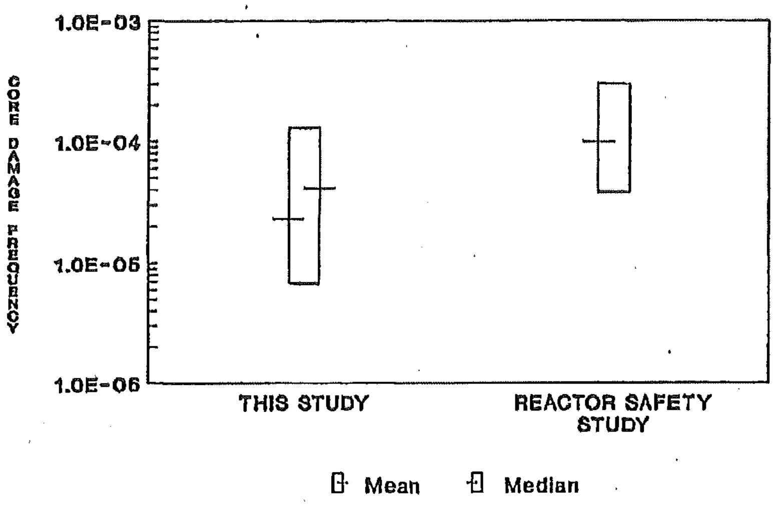
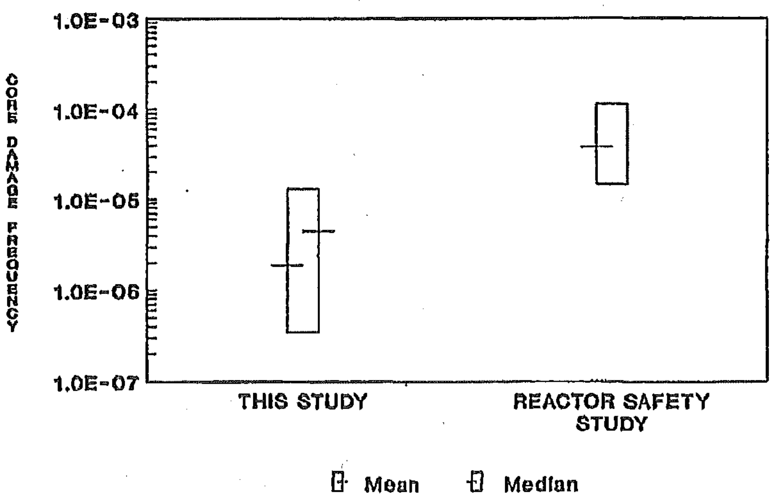


Figure 8.13 Comparison of Surry internal core damage frequency with Reactor Safety Study.



Note: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Figure 8.14 Comparison of Peach Bottom internal core damage frequency with Reactor Safety Study.

recovery of ac power, the Reactor Safety Study did not consider recovery actions). Thus, the net effect of including these new techniques and experience is plant specific and can shift core damage frequencies in either higher or lower directions.

In the case of the Surry analysis, the Reactor Safety Study found the core damage frequency to be dominated by loss-of-coolant accidents (LOCAs). For the present study, station blackout accidents are dominant, while the LOCA-induced core damage frequency is substantially reduced from that of the Reactor Safety Study, particularly for the small LOCA events. This occurred in spite of a tenfold increase in the small LOCA initiating event frequency estimates, which was a result of the inclusion of reactor coolant pump seal failures. One reason for the reduction lies in plant modifications made since the Reactor Safety Study was completed. These modifications allow for the crossconnection of the high-pressure safety injection systems, auxiliary feedwater systems, and refueling water storage tanks between the two units at the Surry site. These crossties provide a reliable alternative for recovery of system failures. Thus, the plant modifications (the crossconnections) have driven the core damage frequencies downward, but new PRA information (the higher small LOCA frequency) has driven them upward. In this case, the net effect is an overall reduction in the core damage frequency for internal events.

In the case of Peach Bottom, the Reactor Safety Study found the core damage frequency to be comprised primarily of ATWS accident sequences and of transients with long-term failure of decay heat removal. The present study concludes that station blackout scenarios are dominant. The possibility of containment venting and allowing for some probability of core cooling after containment failure has considerably reduced the significance of the long-term loss of decay heat removal accidents. In addition, the plant has implemented some ATWS improvements, although ATWS events remain among the dominant accident sequence types. Moreover, more modern neutronic and thermal-hydraulic simulations of the ATWS sequences have calculated lower core power levels during the event, allowing more opportunity for mitigation such as through the use of low-pressure injection systems. Thus, for Peach Bottom, both advances in PRA methodology and plant modifications have contributed to a reduction in the estimated core damage frequency from internal events.

In summary, there have been reductions in the core damage frequencies for both plants since the Reactor Safety Study. The reduction in core damage frequency for Peach Bottom is more significant than for Surry; however, there is still considerable overlap of the uncertainty ranges of the two studies. The conclusion to be drawn is that the hardware and procedural changes made since the Reactor Safety Study appear to have reduced the core damage frequency at these two plants, even when accounting for more accurate failure data and reflecting new sequences not identified in the Reactor Safety Study (e.g., the reactor coolant pump seal LOCA).

8.4 Perspectives

8.4.1 Internal-Event Core Damage Probability Distributions

The core damage frequencies produced by all PRAs inherently have large uncertainties. Therefore, comparisons of frequencies between PRAs or with absolute limits or goals are not simply a matter of comparing two numbers. It is more appropriate to observe how much of the probability distribution lies below a given point, which translates into a measure of the probability that the point has not been exceeded. For example, if the median were exactly equal to the point in question, half of the distribution would lie above and half below the point, and there would be a 50 percent probability that the point had not been exceeded.

Similarly, when comparing core damage frequencies calculated for two or more plants, it is not sufficient to simply compare the mean values of the probability distributions. Instead, one must compare the entire distribution. If one plant's distribution were almost entirely below that of another, then there would be a high probability that the first plant had a lower core damage frequency than the second. Seldom is this the case, however. Usually, the distributions have considerable overlap, and the probability that one plant has a higher or lower core damage frequency than another must be calculated. References 8.1 through 8.5 contain more detailed information on the distributions that would support such calculations.

Although the distributions are not compared in detail here, the overlap of such core damage frequency distributions is clearly shown in Figure 8.1. For example, one can have relatively high confidence that the internal-event core damage frequency for Grand Gulf is lower than that of Sequoyah or Surry. Conversely, it can readily be seen that the differences in core damage frequency between Surry and Sequoyah are not very significant.

Interpretation of extremely low median or mean core damage frequencies (<1E-5) is somewhat difficult. As discussed in Section 1.3 and in Reference 8.7, there are limitations in the scope of the study that could lead to actual core damage frequencies higher than those estimated. In addition, the uncertainties in the sequences included in the study tend to become more important on a relative scale as the frequency decreases. A very low core damage frequency is evident for Grand Gulf with the median of the distribution in the range of 1E-6 per reactor year. However, it is incomplete to simply state that the core damage frequency for this plant is that low since the 95th percentile exceeds 1E-5 per reactor year. Thus, although the central tendency of the calculation is very low, there is still a finite probability of a higher core damage frequency, particularly when considering that the scope of the study does not include certain types of accidents as discussed in Section 1.3.

8.4.2 Principal Contributors to Uncertainty in Core Damage Frequency

In Section 8.4.3, analyses are discussed concerning some of the issues and events that contribute to the magnitude of the core damage frequency. Generally, for the accident frequency analysis, the issues that contribute most to the magnitude of the frequency are also the issues that contribute most to the estimated uncertainty. More detail concerning the contributions of various parameters to the uncertainty in core damage frequency may be found in References 8.1 through 8.5. Perspectives on the contributions of accident frequency issues to the uncertainty in risk may be found in Chapter 12.

8.4.3 Dominant Accident Sequence Types

The various accident sequences that contribute to the total core damage frequency can be grouped by common factors into categories. Older PRAs generally did this in terms of the initiating event, e.g., transient, small LOCA, large LOCA. Current practice also uses categories, such as ATWS, seal LOCA, and station blackout. Generally, these categories are not equal contributors to the total core damage frequency. In practice, four or five sequence categories, sometimes fewer, usually contribute almost all the core damage frequency. These will be referred to below as the dominant plant damage states (PDSs).

It should be noted that the selection of categories is not unique in a mathematical sense, but instead is a convenient way to group the results. If the core damage frequency is to be changed, changing something common to the dominant PDS will have the most effect. Thus, if a particular plant had a relatively high core damage frequency and a particular group of sequences were high, a valuable insight into that plant's safety profile would be obtained.

It should also be noted that the importance of the highest frequency accident sequences should be considered in relationship to the total core damage frequency. The existence of a highly dominant accident sequence or PDS does not of itself imply that a safety problem exists. For example, if a plant already had an extremely low estimated core damage frequency, the existence of a single, dominant PDS would have little significance. Similarly, if a plant were modified such that the dominant PDS were eliminated entirely, the next highest PDS would become the most dominant contributor.

Nevertheless, it is the study of the dominant PDS and the important failures that contribute to those sequences that provides understanding of why the core damage frequency is high or low relative to other plants and desired goals. This qualitative understanding of the core damage frequency is necessary to make practical use of the PRA results and improve the plants, if necessary.

Given this background, the dominant PDSs for the five studies are illustrated in Figures 8.2, 8.3, and 8.4. Additional discussion of these PDSs can be found in Chapters 3 through 7. Several observations on these PDSs and their effects on the core damage frequency can be made, as discussed below.

Boiling Water Reactor versus Pressurized Water Reactor

It is evident from Figure 8.1 that the two particular BWRs in this study have internal-event core damage frequency distributions that are substantially lower than those of the three PWRs. While it would be inappropriate to conclude that all BWRs have lower core damage frequencies than PWRs, it is useful to consider why the core damage frequencies are lower for these particular BWRs.

The LOCA sequences, often dominant in the PWR core damage frequencies, are minor contributors in the case of the BWRs. This is not surprising in view of the fact that most BWRs have many more systems than PWRs for injecting water directly into the reactor coolant system to provide makeup. For BWRs, this includes two low-pressure emergency core cooling (ECC) systems (low-pressure coolant injection and low-pressure core spray), each of which is multitrain; two high-pressure injection systems (reactor core isolation cooling and either high-pressure coolant injection or high-pressure core spray); and usually several other alternative injection systems, such as the control rod drive hydraulic system, condensate, service water, firewater, etc. In contrast, PWRs generally have one high-pressure and one low-pressure ECC system (both multitrain), plus a set of accumulators. The PWR ECCS does have considerable redundancy, but not as much as that of most BWRs.

For many types of transient events, the above arguments also hold. BWRs tend to have more systems that can provide decay heat removal than PWRs. For transient events that lead to loss of water inventory due to stuck-open relief valves or primary system leakage, BWRs have numerous systems to provide makeup. ATWS events and station blackout events, as discussed below, affect both PWRs and BWRs.

BWRs have historically been considered more subject than PWRs to ATWS events. This perception was partly due to the fact that some ATWS events in a BWR involve an insertion of positive reactivity. Except for the infrequent occurrence of an unfavorable moderator temperature coefficient, an ATWS event in a PWR is slower, allowing more time for mitigative action.

In spite of this historical perspective for ATWS, it is evident from Figures 8.2 and 8.3 that the ATWS frequencies for the two BWRs are not dramatically higher than for the PWRs. There are several reasons for this. First, plant procedures for dealing with ATWS events have been modified over the past several years, and operator training specifically for these events has improved significantly. Second, the ability to model and analyze ATWS events has improved. More modern neutronic and thermal-hydraulic simulations of the ATWS sequences have calculated lower core power levels during the event than predicted in the past. Further, these calculations indicate that low-pressure injection systems can be used without resulting in significant power oscillations, thus allowing more opportunity for mitigation. Note that for both BWRs and PWRs the frequency of reactor protection system failure remains highly uncertain. Therefore, all comparisons concerning ATWS should be made with caution.

Station blackout accidents contribute a high percentage of the core damage frequency for the BWRs. However, when viewed on an absolute scale, station blackout has a higher frequency at the PWRs than at the BWRs. To some extent this is due to design differences between BWRs and PWRs leading to different susceptibilities. For example, in station blackout accidents, PWRs are potentially vulnerable to reactor coolant pump seal LOCAs following loss of seal cooling, leading to loss of inventory with no method for providing makeup. BWRs, on the other hand, have at least one injection system that does not require ac power. While important, it would be incorrect to imply that the differences noted above are the only considerations that drive the variations in the core damage frequency. Probably more important is the electric power system design at each plant, which is largely independent of the plant type. The station blackout frequency is low at Peach Bottom because of the presence of four diesels that can be shared between units and a maintenance program that led to an order of magnitude reduction in the diesel generator failure rates. Grand Gulf has essentially three trains of emergency ac power for one unit, with one of the trains being both diverse and independent from the other two. These characteristics of the electric power system design tend to dominate any differences in the reactor design. Therefore, a BWR with a below average electric power system reliability could be expected to have a higher station blackout-induced core damage frequency than a PWR with an above average electric power system.

For both BWRs and PWRs, the analyses indicate that, along with electric power, other support systems, such as service water, are quite important. Because these systems vary considerably among plants, caution must be exercised when making statements about generic classes of plants, such as PWRs versus BWRs. Once significant plant-specific vulnerabilities are removed, support-system-driven sequences will probably dominate the core damage frequency of both types of plants. Both types of plants have sufficient redundancy and diversity so as to make multiple independent failures unlikely. Support system failures introduce dependencies among the systems and thus can become dominant.

Boiling Water Reactor Observations

As shown in Figure 8.1, the internal-event core damage frequencies for Peach Bottom and Grand Gulf are extremely low. Therefore, even though dominant plant damage states and contributing failure events can be identified, these items should not be considered as safety problems for the two plants. In fact, these dominating factors should not be overemphasized because, for core damage frequencies below 1E–5, it is possible that other events outside the scope of these internal-event analyses are the ones that actually dominate. In the cases of these two plants, the real perspectives come not from understanding why particular sequences dominate, but rather why all types of sequences considered in the study have low frequencies for these plants.

Previously it was noted that LOCA sequences can be expected to have low frequencies at BWRs because of the numerous systems available to provide coolant injection. While low for both plants, the frequency of LOCAs is higher for Peach Bottom than for Grand Gulf. This is primarily because Grand Gulf is a BWR–6 design with a motor-driven high-pressure core spray system, rather than a steam-driven high-pressure coolant injection system as is Peach Bottom. Motor-driven systems are typically more reliable than steam-driven systems and, more importantly, can operate over the entire range of pressures experienced in a LOCA sequence.

It is evident from Figures 8.2 and 8.4 that station blackout plays a major role in the internal-event core damage frequencies for Peach Bottom and Grand Gulf. Each of these plants has features that tend to reduce the station blackout frequency, some of which would not be present at other BWRs.

Grand Gulf, like all BWR–6 plants, is equipped with an extra diesel generator dedicated to the high-pressure core spray system. While effectively providing a third train of redundant emergency ac power for decay heat removal, the extra diesel also provides diversity, based on a different diesel design and plant location relative to the other two diesels. Because of the aspect of diversity, the analysis neglected common-cause failures affecting all three diesel generators. The net effect is a highly reliable emergency ac power capability. In those unlikely cases where all three diesel generators fail, Grand Gulf relies on a steam-driven coolant injection system that can function until the station batteries are depleted. At Grand Gulf the batteries are sized to last for many hours prior to depletion so that there is a high probability of recovering ac power prior to core damage. In addition, there is a diesel-driven firewater system available that can be used to provide coolant injection in some sequences involving the loss of ac power.

Peach Bottom is an older model BWR that does not have a diverse diesel generator for the high-pressure core spray system. However, other factors contribute to a low station blackout frequency at Peach Bottom. Peach Bottom is a two-unit site, with four diesel generators available. Any one of the four diesels can provide sufficient capacity to power both units in the event of a loss of offsite power, given that appropriate crossties or load swapping between Units 2 and 3 are used. This high level of redundancy is somewhat offset by a less redundant service water system that provides cooling to the diesel generators. Subtleties in the design are such that if a certain combination of diesel generators fails, the service water system will fail, causing the other diesels to fail. In addition, station dc power is needed to start the diesels. (Some emergency diesel generator systems, such as those at Surry, have a separate dedicated dc power system just for starting purposes.) In spite of these factors, the redundancy in the Peach Bottom emergency ac power system is considerable.

While there is redundancy in the ac power system design at Peach Bottom, the most significant factor in the low estimated station blackout frequency relates to the plant-specific data analysis. The plant-specific analysis determined that, because of a high-quality maintenance program, the diesel generators at Peach Bottom had approximately an order of magnitude greater reliability than at an average plant. This factor directly influences the frequency.

Finally, Peach Bottom, like Grand Gulf, has station batteries that are sized to last several hours in the event that the diesel generators do fail. With two steam-driven systems to provide coolant injection and several hours to recover ac power prior to battery depletion, the station blackout frequency is further reduced.

Unlike most PWRs, the response of containment is often a key in determining the core damage frequency for BWRs. For example, at Peach Bottom, there are a number of ways in which containment conditions can affect coolant injection systems. High pressure in containment can lead to closure of primary system relief valves, thus failing low-pressure injection systems, and can also lead to failure of steam-driven high-pressure injection systems due to high turbine exhaust backpressure. High suppression pool temperatures can also lead to the failure of systems that are recirculating water from the suppression pool to the reactor coolant system. If the containment ultimately fails, certain systems can fail because of the loss of net positive suction head in the suppression pool, and also the reactor building is subjected to a harsh steam environment that can lead to failure of equipment located there.

Despite the concerns described in the previous paragraph, the core damage frequency for Peach Bottom is relatively low, compared to the PWRs. There are two major reasons for this. First, Peach Bottom has the ability to vent the wetwell through a 6-inch diameter steel pipe, thus reducing the containment pressure without subjecting the reactor building to steam. While this vent cannot be used to mitigate ATWS and station blackout sequences, it is valuable in reducing the frequency of many other sequences. The second important feature at Peach Bottom is the presence of the control rod drive system, which is not affected by either high pressure in containment or containment failure. Other plants of the BWR-4 design may be more susceptible to containment-related problems if they do not have similar features. For example, some plants have ducting, as opposed to hard piping available for venting. Venting through ductwork may lead to harsh steam environments and equipment failures in the reactor building.*

The Grand Gulf design is generally much less susceptible to containment-related problems than Peach Bottom. The containment design and equipment locations are such that containment rupture will not result in discharge of steam into the building containing the safety systems. Further, the high-pressure core spray system is designed to function with a saturated suppression pool so that it is not affected by containment failure. Finally, there are other systems that can provide coolant injection using water sources other than the suppression pool. Thus, containment failure is relatively benign as far as system operation is concerned, and there is no obvious need for containment venting.

Pressurized Water Reactor Observations

The three PWRs examined in this study reflect much more variety in terms of dominant plant damage states than the BWRs. While the sequence frequencies are generally low for most of the plant damage states, it is useful to understand why the variations among the plants occurred.

For LOCA sequences, the frequency is significantly lower at Surry than at the other two PWRs. A major portion of this difference is directly tied to the additional redundancy available in the injection systems. In addition to the normal high-pressure injection capability, Surry can crosstie to the other unit at the site for an additional source of high-pressure injection. This reduces the core damage frequency due to LOCAs and also certain groups of transients involving stuck-open relief valves.

In addition, at Sequoyah there is a particularly noteworthy emergency core cooling interaction with containment engineered safety features in loss-of-coolant accidents. In this (ice condenser) containment design, the containment sprays are automatically actuated at a very low pressure setpoint, which would be exceeded for virtually all small LOCA events. This spray actuation, if not terminated by the operator can lead to a rapid depletion of the refueling water storage tank at Sequoyah. Thus, an early need to switch to recirculation cooling may occur. Portions of this switchover process are manual at Sequoyah and, because of the timing and possible stressful conditions, leads to a significant human error probability. Thus, LOCA-type sequences are the dominant accident sequence type at Sequoyah.

Station blackout-type sequences have relatively similar frequencies at all three PWRs. Station blackout sequences can have very different characteristics at PWRs than at BWRs. One of the most important findings of the study is the importance of reactor coolant pump seal failures. During station blackout, all cooling to the seals is lost and there is a significant probability that they will ultimately fail, leading to an induced LOCA and loss of inventory. Because PWRs do not have systems capable of providing coolant makeup without ac power, core damage will result if power is not restored. The seal LOCA reduces the time available to restore power and thus increases the station blackout-induced core damage frequency. New seals have been proposed for Westinghouse PWRs and could reduce the core damage frequency if implemented, although they might also increase the likelihood that any resulting accidents would occur at high pressure, which has implications for the accident progression analysis. (See Section C.14 of Appendix C for a more detailed discussion of reactor coolant seal performance.)

Apart from the generic reactor coolant pump seal question, station blackout frequencies at PWRs are determined by the plant-specific electric power system design and the design of other support systems. Battery depletion times for the three PWRs were projected to be shorter than for the two BWRs. A particular characteristic of the

*The staff is presently undertaking regulatory action to require hard pipe vents in all BWR Mark I plants.

Surry plant is a gravity-fed service water system with a canal that may drain during station blackout, thus failing containment heat removal. When power is restored, the canal must be refilled before containment heat removal can be restored.

The dominant accident sequence type at Zion is not a station blackout, but it has many similar characteristics. Component cooling water is needed for operation of the charging pumps and high-pressure safety injection pumps at Zion. Loss of component cooling water (or loss of service water, which will also render component cooling water inoperable) will result in loss of these high-pressure systems. This in turn leads to a loss of reactor coolant pump seal injection. Simultaneously, loss of component cooling water will also result in loss of cooling to the thermal barrier heat exchangers for the reactor coolant pump seals. Thus, the reactor coolant pump seals will lose both forms of cooling. As with station blackout, loss of component cooling water or service water can both cause a small LOCA (by seal failure) and disable the systems needed to mitigate it. The importance of this scenario is increased further by the fact that the component cooling water system at Zion, although it uses redundant pumps and valves, delivers its flow through a common header. The licensee for the Zion plant has made procedural changes and is also considering both the use of new seal materials and the installation of modifications to the cooling water systems. These measures, which are discussed in more detail in Chapter 7, reduce the importance of this contributor.

ATWS frequencies are generally low at all three of the PWRs. This is due to the assessed reliability of the shutdown systems and the likelihood that only slow-acting, low-power-level events will result.

While of low frequency, it is worth noting that interfacing-system LOCA (V) and steam generator tube rupture (SGTR) events do contribute significantly to risk for the PWRs. This is because they involve a direct path for fission products to bypass containment. There are large uncertainties in the analyses of these two accident types, but these events can be important to risk even at frequencies that may be one or two orders of magnitude lower than other sequence types.

During the past few years, most Westinghouse PWRs have developed procedures for using feed and bleed cooling and secondary system blowdown to cope with loss of all feedwater. These procedures have led to substantial reductions in the frequencies of transient sequences involving the loss of main and auxiliary feedwater. Appropriate credit for these actions was given in these analyses. However, there are plant-specific features that will affect the success rate of such actions. For example, the loss of certain power sources (possibly only one bus) or other support systems can fail power-operated relief valves (PORVs) or atmospheric dump valves or their block valves at some plants, precluding the use of feed and bleed or secondary system blowdown. Plants with PORVs that tend to leak may operate for significant periods of time with the block valves closed, thus making feed and bleed less reliable. On the other hand, if certain power failures are such that open block valves cannot be closed, then they cannot be used to mitigate stuck-open PORVs. Thus, both the system design and plant operating practices can be important to the reliability assessment of actions such as feed and bleed cooling.

8.4.4 External Events

The frequency of core damage initiated by external events has been analyzed for two of the plants in this study, Surry and Peach Bottom (Ref. 8.1 (Part 3) and Ref. 8.2 (Part 3)). The analysis examined a broad range of external events, e.g., lightning, aircraft impact, tornados, and volcanic activity (Ref. 8.8). Most of these events were assessed to be insignificant contributors by means of bounding analyses. However, seismic events and fires were found to be potentially major contributors and thus were analyzed in detail.

Figures 8.7 and 8.8 show the results of the core damage frequency analysis for seismic- and fire-initiated accidents, as well as internally initiated accidents, for Surry and Peach Bottom, respectively. Examination of these figures shows that the core damage frequency distributions of the external events are comparable to those of the internal events. It is evident that the external events are significant in the total safety profile of these plants.

Seismic Analysis Observations

The analysis of the seismically induced core damage frequency begins with the estimation of the seismic hazard, that is, the likelihood of exceeding different earthquake ground-motion levels at the plant site. This is a difficult, highly judgmental issue, with little data to provide verification of the various proposed geologic and seismologic models.

The sciences of geology and seismology have not yet produced a model or group of models upon which all experts agree. This study did not itself produce seismic hazard curves, but instead made use of seismic hazard curves for Peach Bottom and Surry that were part of an NRC-funded Lawrence Livermore National Laboratory project that resulted in seismic hazard curves for all nuclear power plant sites east of the Rocky Mountains (Ref. 8.9).

In addition, the Electric Power Research Institute (EPRI) developed a separate set of models (Ref. 8.10). For purposes of completeness and comparison, the seismically induced core damage frequencies were also calculated based upon the EPRI methods. Both sets of results, which are presented in Figures 8.5 through 8.8, were used in this study. More detailed discussion of methods used in the seismic analysis is provided in Appendix A; Section C.11 of Appendix C provides more detailed perspectives on the seismic issue as well.

As can be seen in Figures 8.5 and 8.6, the shapes of the seismically induced core damage probability distributions are considerably different from those of the internally initiated and fire-initiated events. In particular, the 5th to 95th percentile range is much larger for the seismic events. In addition, as can be seen in Figures 8.7 and 8.8, the wide disparity between the mean and the median and the location of the mean relatively high in the distribution indicate a wide distribution with a tail at the high end but peaked much lower down. (This is a result of the uncertainty in the seismic hazard curve.)

It can be clearly seen that the difference between the mean and median is an important distinction. The mean is the parameter quoted most often, but the bulk of the distribution is well below the mean. Thus, although the mean is the "center of gravity" of the distribution (when viewed on a linear rather than logarithmic scale), it is not very representative of the distribution as a whole. Instead, it is the lower values that are more probable. The higher values are estimated to have low probability, but, because of their great distance from the bulk of the distribution, the mean is "pulled up" to a relatively high value. In a case such as this, it is particularly evident that the entire distribution, not just a single parameter such as the mean or the median, must be considered when discussing the results of the analysis.

1. Surry Seismic Analysis

The core damage frequency probability distributions, as calculated using the Livermore and EPRI methods, have a large degree of overlap, and the differences between the means and medians of the two resulting distributions are not very meaningful because of the large widths of the two distributions.

The breakdown of the Surry seismic analysis into principal contributors is reasonably similar to the results of other seismic PRAs for other PWRs. The total core damage frequency is dominated by loss of offsite power transients resulting from seismically induced failures of the ceramic insulators in the switchyard. This dominant contribution of ceramic insulator failures has been found in virtually all seismic PRAs to date.

A site-specific but significant contributor to the core damage frequency at Surry is failure of the anchorage welds of the 4 kV buses. These buses play a vital role in providing emergency ac electrical power since offsite power as well as emergency onsite power passes through these buses. Although these welded anchorages have more than adequate capacity at the safe shutdown earthquake (SSE) level, they do not have sufficient margin to withstand (with high reliability) earthquakes in the range of four times the SSE, which are contributing to the overall seismic core damage frequency results.

Similarly, a substantial contribution is associated with failures of the diesel generators and associated load center anchorage failures. These anchorages also may not have sufficient capacity to withstand earthquakes at levels of four times the SSE.

Another area of generic interest is the contribution due to vertical flat-bottomed storage tanks, e.g., refueling water storage tanks and condensate storage tanks. Because of the nature of their configuration and field erection practices, such tanks have often been calculated to have relatively smaller margin over the SSE than most components in commercial nuclear power plants. Given that all PWRs in the United States use the refueling water storage tank as the primary source of emergency injection water (and usually the sole source until the recirculation phase of ECCS begins), failure of the refueling water storage tank can be expected to be a substantial contributor to the seismically induced core damage frequency.

2. Peach Bottom Seismic Analysis

As can be seen in Figure 8.9, the dominant contributor in the seismic core damage frequency analysis is a transient sequence brought about by loss of offsite power. The loss of offsite power is due to seismically induced failures of onsite ac power. Peach Bottom has four emergency diesel generators, all shared between the two units, and four station batteries per unit. Thus, there is a high degree of redundancy. However, all diesels require cooling provided by the emergency service water system, and failure to provide this cooling will result in failure of all four diesels.

There is a variety of seismically induced equipment failures that can fail the emergency service water system and result in a station blackout. These include failure of the emergency cooling tower, failures of the 4 kV buses (in the same manner as was found at Surry), and failures of the emergency service water pumps or the emergency diesel generators themselves. The various combinations of these failures result in a large number of potential failure modes and give rise to a relatively high frequency of core damage based on station blackout. None of these equipment failure probabilities is substantially greater than would be implied by the generic fragility data available. However, the high probability of exceedance of larger earthquakes (as prescribed by the hazard curves for this site) results in significant contributions of these components to the seismic risk.

Fire Analysis Observations

The core damage likelihood due to a fire in any particular area of the plant depends upon the frequency of ignition of a fire in the area, the amount and nature of combustible material in that area, the nature and efficacy of the fire-suppression systems in that area, and the importance of the equipment located in that area, as expressed in the potential of the loss of that equipment to cause a core damage accident sequence. The methods used in the fire analysis are described in Appendix A and in Reference 8.7; Section C.12 of Appendix C provides additional perspectives on the fire analysis.

1. Surry Fire Analysis

Figure 8.10 shows the dominant contributors to core damage frequency resulting from the Surry fire analysis. The dominant contributor is a transient resulting in a reactor coolant pump seal LOCA, which can lead to core damage. The scenario consists of a fire in the emergency switchgear room that damages power or control cables for the high-pressure injection and component cooling water pumps. No additional random failures are required for this scenario to lead to core damage. It should be noted that credit was given for existing fire-suppression systems and for recovery by crossconnecting high-pressure injection from the other unit. The importance of this scenario is evident in Figure 8.11, which breaks down the fire-induced core damage frequency by location in the plant. The most significant physical location is the emergency switchgear room. In this room, cable trays for the two redundant power trains were run one on top of the other with approximately 8 inches of vertical separation in a number of plant areas, which gives rise to the common vulnerability of these two systems due to fire. In addition, the Halon fire-suppression system in this room is manually actuated.

The other principal contributor is a spuriously actuated pressurizer PORV. In this scenario, fire-related component damage in the control room includes control power for a number of safety systems. Full credit was given for independence of the remote shutdown panel from the control room except in the case of PORV block valves; discussions with utility personnel indicated that control power for these valves was not independently routed.

2. Peach Bottom Fire Analysis

Figure 8.10 shows the mechanisms by which fire leads to core damage in the Peach Bottom analysis. Station blackout accidents are the dominant contributor, with substantial contributions also coming from fire-induced transients and losses of offsite power. The relative importance of the various physical locations is shown in Figure 8.12.

It is evident from Figure 8.12 that control room fires are of considerable significance in the fire analysis of this plant. Fires in the control room were divided into two scenarios, one for fires initiating in the reactor core isolation cooling (RCIC) system cabinet and one for all others. Credit was given for automatic cycling of the RCIC system unless the fire initiated within its control panel. Because of the cabinet configuration within the control room, the fire was assumed not to spread and damage any components outside the cabinet where the fire initiated. The analysis gave credit for the possibility of quick extinguishing of the fire within the applicable cabinet since the control room is continuously occupied. However, should these efforts fail, even with high ventilation rates, these scenarios postulate forced abandonment of the control room due to smoke from the fire and subsequent plant control from the remote shutdown panel.

The cable spreading room below the control room is significant but not dominant in the fire analysis. The scenario of interest is a fire-induced transient coupled with fire-related failures of the control power for the high-pressure coolant injection system, the reactor core isolation cooling system, the automatic depressurization system, and the control rod drive hydraulic system. The analysis gave credit to the automatic CO_2 fire-suppression system in this area.

The remaining physical areas of significance are the emergency switchgear rooms. The fire-induced core damage frequency is dominated by fire damage to the emergency service water system in conjunction with random failures coupled with fire-induced loss of offsite power. In all eight emergency switchgear rooms (four shared between the two units), both trains of offsite power are routed. It was noted that in each of these areas there are breaker cubicles for the 4 kV switchgear with a penetration at the top that has many small cables routed through it. These penetrations were inadequately sealed, which would allow a fire to spread to cabling that was directly above the switchgear room. This cabling was a sufficient fuel source for the fire to cause a rapid formation of a hot gas layer that would then lead to a loss of offsite power. Since both offsite power and the emergency service water systems are lost, a station blackout would occur.

Perspectives: General Observations on Fire Analysis

Figures 8.7 and 8.8 clearly indicate that fire-initiated core damage sequences are significant in the total probabilistic analysis of the two plants analyzed. Moreover, these analyses already include credit for the fire protection programs required by Appendix R to 10 CFR Part 50.

Although the two plants are of completely different design, with completely different fire-initiated core damage scenarios, the possibility of fires in the emergency switchgear areas is important in both plants. The importance of the emergency switchgear room at Surry is particularly high because of the seal LOCA scenario. Further, the importance of the control room at Surry is comparable to that of the control room at Peach Bottom.

This is not surprising in view of the potential for simultaneous failure of several systems by fires in these areas. Thus, in the past such areas have generally received particular attention in fire protection programs. It should also be noted that the significance of various areas also depends upon the scenario that leads to core damage. For example, the importance of the emergency switchgear room at Surry could be altered (if desired) not only by more fire protection programs but also by changes in the probability of the reactor coolant pump seal failure.

東電福島第一事故について

近藤駿介

原子炉安全の歴史

- 人里から出力に応じた適当な離隔距離Rをとれるところに立地すること $R^2 \sim P$　物理的安全
- 原子炉暴走時の発生エネルギーは負の反応度フィードバックを備えることにより制限できる→格納容器の導入で離隔距離を短くすることを容認:原子力発電の始まり
- 格納容器は炉心溶融時に機能するか？→緊急時炉心冷却システム（ECCS）の設置
- こうした工学的安全施設はどこまで信頼できるものであるべきか→How safe is safe enough? 信頼性論争の始まり;
  - 深層防護に基づく安全確保方針:高信頼設計の実施、設計基準事故・事象に備える工学的安全設備、防災計画の整備
  - リスク評価の実施:
    - 早期放出の防止、フィルター付き格納容器ベントの活用等のシビアアクシデントのマネジメントも効果的
    - 安全目標（世界に1000基の原子炉があるとして、これの大事故は、人がそれについて一生のうちに一度きくことがある程度まれなものにすべき）の達成には、
      - 設計基準事象には再帰期間が10000年の事象強度を選ぶこと
      - 設計に当たっては事象強度の不確実性が大きいことを念頭に、強度の不確実性に対して感度が低いようにすること

Reflections on Fukushima

19th International Conference On Nuclear Engineering (ICONE19) , October 24-25, 2011 Osaka, Japan

Dr. Nils J. Diaz

Managing Director, The ND2 Group, LLC Chairman, ASME Task Force- Japan Events

Nuclear Power Before Fukushima

- 25 years without a nuclear reactor accident provided assurance of safety worldwide. The two prior accidents:
  - TMI: Light Water Reactor (LWR), caused by internal event (open valve) and human factors, resulting in core degradation but no public health & safety/environmental effects.
  - Chernobyl: Graphite Reactor, caused by internal event, major errors in operation (human factors), resulting in core melting/burning, uncontrolled radioactive releases with major public health & safety and environmental consequences.
- No LWR accident until Fukushima had resulted in significant external radioactive contamination.
- No external event before Fukushima resulted in core degradation and uncontrolled radioactivity release (including Armenia).
- Regardless of initiating events, all accidents resulted in loss of core cooling after deficient accident management and human errors.

Reflections on Fukushima

19th International Conference On Nuclear Engineering (ICONE19) , October 24-25, 2011 Osaka, Japan
Dr. Nils J. Diaz
Managing Director, The ND2 Group, LLC Chairman, ASME Task Force- Japan Events

The Fukushima Daiichi Nuclear Plant Accidents

What should have happened?

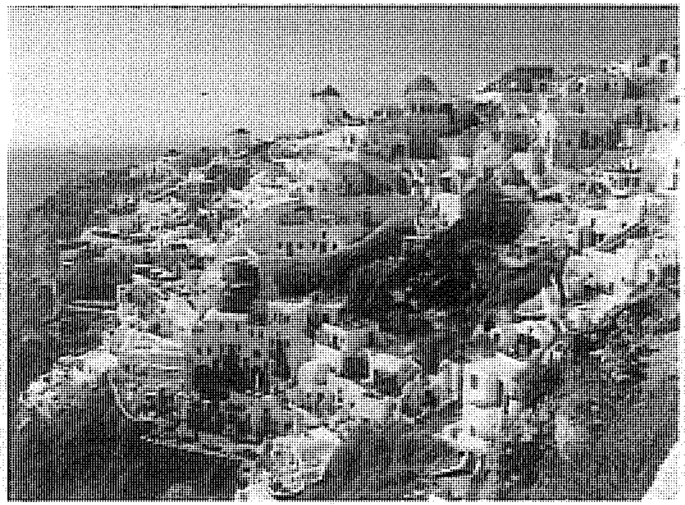
- Provide adequate heat removal for the reactor core and safety- related heat sources even during accident conditions.
  - The above dominant reactor safety criteria is met by: maintaining core cooling; maintaining cooling of spent fuel pools; maintaining containment integrity; maintaining command and control; effective accident management complemented by emergency preparedness to prevent or minimize radiological releases to the public and the environment.

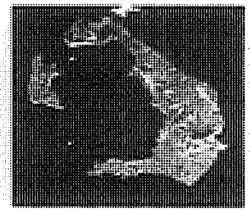
What happened?

- *Loss of power resulted in lack of heat removal and core degradation.*
- Multi-unit reactor accidents, most complex nuclear power scenarios
- ever.
- A major environmental radiological contamination, apparently without serious radiological public health and safety consequences.

事故は何故防げなかったのか

−東大原子力GCOEによる「ご意見を聞く会」から−



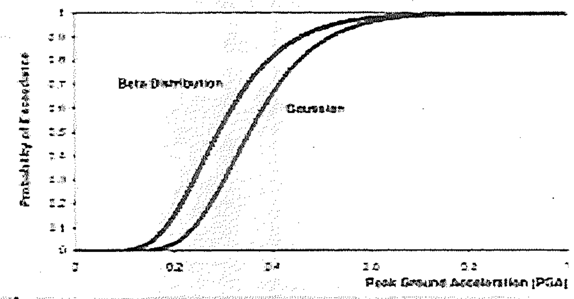


東大原子力GCOE	特任教授	尾本　彰
	特任助教	寿楽　浩太
東大原子力国際専攻	教授	田中　知

尾本、寿楽、田中

自然災害への備えの不足

- 原子力安全をリードした専門家の間で、地震については危機感を持って検討がされたが、津波への警戒心が不足していた
- 地震に対して持つ工学的余裕特性と津波のそれとの相違に注意が必要だった（Fragility curve)



- 米国設計を墨守しすぎて日本固有の気候風土（地震津波火山)、社会（沿岸部の稠密利用、土地汚染の重要性）への配慮不足
- 内因事象検討に力点が置かれ、外因事象とテロ対策に遅れ、かつ、外因事象に関する情報源（認識科学）との間に隙間
- 理学における不確かさへの認識不足。工学的余裕を含め理学/工学の議論不足
- 自然災害（外因事象）への安全目標の適用が理解されていなかった
- 今回の津波は「合理的な予防行動」（十分な説得性をもって専門家間で予防行動を取るべきとの合意を得ての行動）には至らないであったろうが、残余のリスクにどう対処するかの視点が不足していた
- 5年前には気づいたとしても，その前の40年間にもリスクはあった。偶然今年起った。そう考えると、そもそも防げなかったのではないか？

尾本、寿楽、田中

規制

- 事故は規制システムの欠陥によっておきた
  - ✓新しい科学的知見を法規制に反映する仕組みの不足と遅れ
  - ✓シビアアクシデントを含め規制の国際整合への立ち遅れ（2002年の東電問題で別の課題解決に奔走など）
  - ✓規制一貫化前の2段階規制の後遺症として、構造材料規制偏重
  - ✓IAEA基本安全原則に忠実でなく、安全目標も無い独自の規制
  - ✓安全委員会と保安院の責任範囲分解が不明瞭、不適切(例：基準策定箇所と基準を用いて審査する箇所の分離)
  - ✓実プラントの設計や運転の判る規制当局者が居ない（専門能力涵養に適さない役人の人事異動システム)
  - ✓ 規制の過剰な品質保証重視行政によって発電所員が書類作成に追われ現場にも行けず安全への問題意識が希薄に
- 貞観地震津波をきちんと受け止めず適切な措置を取らなかった
- 事業者/規制ともに、安全文化と責任感と敏感さが不足
- 研究者には意見を求めてもプラントを知る事業者に意見を求めず
- 事業者と大学等の専門家が個別許認可に関して間接話法
- 規制に世界の風を感じつつ世界に発信する能力が不足

尾本、寿楽、田中

安全問題への対処姿勢

（<u>事業者の安全文化</u>）

➢事業者/規制ともに、安全文化と責任感と敏感さが不足

➢安全文化の劣化
  - ✓発電所長に事務系職員を任命など地元重視の一方で、技術と安全を軽視
  - ✓運転員以外には運転の知識が不足
  - ✓SBOルールやB5bなど海外の動向への敏感さ/学ぶ姿勢不足

➢事前に考えることなく、何か起きないと対処しない姿勢（例：免震重要棟は柏崎刈羽の2007年地震の経験を踏まえたものだが、もしその経験が無かったらどうなっていたか）

➢2007年地震で安全機能が確保された経験から自然災害に対して「安全だ」との思い込み/過信/慢心

➢電力に「規制要求への対応＝安全確保」との誤解があったのでは

➢継続的な改善を行いにくい社会環境（「安全か否か」の二元論、「安全問題があるなら改造完了迄止めるべき」との地元の声）の中で安全問題への対処の遅れ

尾本、寿楽、田中

安全問題への対処姿勢(…続)

(日常に埋没)

- LPHCリスクに鈍感になって行った可能性。日々起こる些細な問題の処理にかまけて、時間を掛け深く考えるべき安全問題が後回しに
- 様々な疑問を封じ込めた日常管理
- 新しい設計に挑戦する機会が減り、ルーチンの運転保守主体の発電所運営で設計や潜在的安全問題への関心低下

(安全専門家)

- 近年、安全専門家が居なくなった
- 真の意味の深層防護(防護手段を尽くしてもなお、その手段を無効にする事象は起こりうるとする思想)が理解されていなかった
- 領域横断型(航空/鉄道/化学)の安全専門家の不在
- 欧米における安全論(技術と社会との係りの議論など)からの隔絶
- 確率を重視し、事故の結果(とりわけ土地汚染)への配慮不足
- 内因事象(TMI, Chernobyl)を重視し外因事象がなおざりに
- 外因事象に対し内因事象と同じ安全目標値を適用するとの合意が不十分

(安全研究)

国の安全研究から軽水炉分野を除外したのは科学技術政策の誤り

尾本、寿楽、田中

原子力界の在り方：社会とのインターフェイス、文化

(責任感)

- 組織を超えて原子力技術者個人が社会に対し有する責任が欠落
- 原子力を担う組織はそれぞれ巨大で、原子力技術者はその巨大組織の一部としてしか能力を発揮できない。その巨大組織は大きすぎて身動きが取れなくなってきているのでは

(自分の立場に囚われすぎて)

- 外部からの批判に無関心または反発(排他性)
- 推進と反対のどちらも自分の立場に囚われすぎて柔軟にリスク低減問題に対処できない

- リスク管理と低減のための活動が堂々と出来ない。当該問題の処理完了まで停止を求められる恐れ等を考え、問題を封印する傾向

(国策推進体制)

- 国策民営の中で、国の権威を利用し産官学が一致して原子力を進める中、疑問を提示しにくい環境、率直な議論がしにくい環境の形成
- 社会的合意形成不足なまま、国と自治体の役割分担不明な儘推進

尾本、寿楽、田中

原子力界の在り方：社会とのインターフェイス、文化(…続)

(<u>社会とのインターフェイス</u>)

- 今回の事故は原子力と社会とのインターフェイスの失敗を意味
  - ✓工学の社会的使命(社会的価値の創造)を考えた社会との対話に欠け、社会の要請を受け止めていない
  - ✓社会と接点をもつ防災に関しての備えが弱かった
- 技術が社会とともに進化するのに原子力は社会リテラシーが不足し硬直的で進化がない
- なぜ社会が受容できないリスクを我々が受容してきたのか？
- 安全・安心という用語に問題(安全は「安心」していては成立せず。が、「そんな事言ったら、日本で原子力は成り立たない」との反論すらある)

(<u>電力の体質</u>)

- 電力自由化のなか、経済性重視へと電力の経営姿勢が変化
- 問題意識が上にゆかない、上意下達が不十分な電力の組織風土
- 組織の大きな慣性力ゆえ、機敏な対応ができない
- 電力では、アウトソースが多く、トラブル対応で技術者は言い訳に長け、形式重視の品質保証の業務が規制に求められデスクワークばかりで現場に密着した仕事をしていない
- 電力経営のリスクに対する注意が立地地域との関係ばかりに向けられてきた

尾本、寿楽、田中

原子力界の在り方：社会とのインターフェイス、文化(…続)

(その他)

- 実学を経験した技術者が減った。計算機のsimulationに没頭し実際を知らない、油にまみれず現場に対処できない技術者が増え、結果として事故時の現実感に基づく対策が不足
- 耳学問でやってきた初期の原子力が旨くゆきすぎた
- 斯界の権威ある専門家によって方向付けがされて他の専門家が抗えない環境(例として、安全目標検討の経緯)
- 多様な意見を議論することが不足、否定的な見解をじっくり議論することが不足した「なれ合いの社会」
- コミュニケーションを得意とする人が，原子力技術者が社会と触れる機会を結果的に奪っているのではないか？
- 「継続的な改善」がなされない(理由：規制産業)、規制を変える声が電力から弱い
- 個人の責任を問うシステムが無い、それをつくる以前に国策民営という巨大な流れができ、考え方の差異の小さい「個人としての自立」がない専門家の集まりとなっている
- 原子力界は国民に向き合ったのか？

尾本、寿楽、田中

安全文化の劣化等を認識していたなら事故要因の形成を防ぐ行動は取られたのか、取られなかったのなら何故取られなかったのか？

（十分に取られなかったのではないか、という立場から）

- 電力には何か起きないと対処しない姿勢がある
- 電力は殿様。とりわけ東電はBWRメーカー２者を国内に抱え、彼らを競争させている。一方、PWRメーカーは１社だから競争がなく、電力にダメですと言える
- 企業の希薄な自己責任意識
- 国策民営で国が方向を規定し産官学が共同して実行するなか、疑問を挟む余地なし
- 組織内で提言/行動への諦め。言われた/定められた事しかしない一般社会の気風に通じる
- 煩雑化する業務の中で思考停止に陥っていたのでは？
- 電力会社で不始末・事故・不祥事ゆえの責任者の更迭が度重なり、リーダーシップを持った人材不足に至ったのでは

尾本、寿楽、田中

事故の拡大要因としての危機管理と防災の問題

- 危機管理に際しての指揮命令系統/指揮者の能力の問題
- 官邸災害対策本部における規制行政の役割が不明瞭不適切で能力の発揮ができなかった
- 危機管理における責任分解点が不明で、事態への対処遅れと混乱
- 安全設計はあっても防災がおろそかであった（この狭い国土で絶対事故を起こしちゃいけないと言うのが第一世代の安全屋にあまりにも強すぎて、安全設計の方に重点を置きすぎた）
- 修羅場の話を真剣にやってきていない
- 本当にアクシデントに拡大していく段階のところで、もっとましな対応ができたんではないかという思いがある
- ロシアの人は、何故自衛隊を出して早期に除染しないのかと言っている。危機管理に問題
- 原子力村の問題と規定するよりも社会全体のリスクガバナンスの問題として考えるべき

尾本、寿楽、田中

その他関連のある資料

<u>矢川元基</u>(原子力学会誌巻頭言Vol53. No.10)

- 欧米原子力先進国では多くの基礎研究を積み重ねながら、時には失敗もしながら知識を蓄えその集大成として原子力利用技術を獲得。一方、わが国では逆の道を辿ってきた。
- 日本人特有の器用さで物づくりとしては世界一の製品を作ってきたかも知れないが、どこかにまだ借り物技術のひ弱さが見られないだろうか

<u>D. Klein</u> (前USNRC委員長, Ripon Forum, Summer 2011)

- The LL from Fukushima are many, but what may be surprising is how few may actually apply to US plants.
- In a culture where it is impolite to say "no" and where ritual must be observed before all else, I think that Western style "safety culture" will be very hard for the Japanese to accept. But accept it they must if they want to achieve excellence.

尾本、寿楽、田中

その他関連資料

原子力総合シンポジウム2011Oct

1．事故の遠因（東大澤田）

✓構造強度偏重
✓省庁間の連携不足で安全確保に国が真剣に取り組まず
✓品質問題に拘り、大局を見失う
✓スクラム頻度などからくる安全への過信
✓基準改定の困難さが新技術を拒絶
✓責任所在不明（審議会委員会での決定の中、責任者が不明）
✓専門化不在の規制

2．何故、M9の地震が予測できなかったのか（入倉）

M9の地震はプレートの沈み込み速度が速い所で発生（チリ型）し、移動速度が遅い（マリアナ型）所はM7級。東北太平洋沖は中間でM8→スマトラ地震（M9.1,マリアナ型）は移動速度が遅い所で発生し、従来の考えは疑問視され、すべり残しによる700年に一回のスーパーサイクルモデル提示(2008, 2011)→地震本部は2002年の「三陸沖から房総沖の地震活動の長期評価」の改訂準備中であった

尾本、寿楽、田中

得られた主要な意見の纏めと要因を推察する過程

この規模の地震津波を地震本部を含め想定できていなかったが、原子力災害は防げた/防ぐべきだった →

（拡大を含め）何故、事故を防げなかったのか？

何故津波への感度が低かったのか？

- ➢自然災 の認識不足
- ➢情報への感受性不足
- ➢米国設計を墨守し、気候（自然災害）/社会（土地汚染の重要性）の相違への配慮が不足
- ➢理学における不確かさへの認識不足。工学的余裕を含め理学/工学の議論不足

設計ベースを超える事故への対策が不十分あるいは機能しなかった背景は？

- ➢事 安全文化劣化（注意深さ欠如、慢心、情報への感受性不足、安全との思い込み、行動遅さ等）
- ➢規制が機能せずリスク放置、規制/事業者の緊張が欠如
- ➢目の前の課題解決に奔走し消耗し安全問題は後回し
- ➢国/事業者の危機管理体制不備
- ➢追加安全対策実施は安全でない証拠→完成迄停止要求
- ➢なぜ社会が受容できないリスクを我々が受容してきたか？

背景に

それを認識していたなら何故実効性ある策が提起されorとられなかったのか？

- ➢世界の動向への関心の薄さ
- ➢閉鎖的集団を形成し社会と弱い繋がり
- ➢誰も責任を取らない無責任体制
- ➢工学が社会の要求に答えていない

今回の事故が無くても抱えていたであろう問題の指摘

- ➢実施したが有効性不足
- ➢電力は殿様
- ➢国策民営で疑問を挟む余地なし
- ➢組織内提言/行動の実効性への諦め
- ➢煩雑化する業務の中で思考停止

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今迄の意見聴取からの洞察

1.「予見が難しくとも対策されるべきだった」との見方が強い
未曾有の地震津波を地震本部も「想定外」としたように予見し難く、「何故防げなかったか」という設問は不適切とする意見はなかった

2.長年の原子力界での活動により得られた知見に立った洞察とはいえ、因果関係の検証を伴わないと、憶測に留まり根本原因は解明されない可能性があるとの問題を感じた

3.「何故、津波への感度が低かったのか」という分析よりも、「事業者の安全文化の劣化」「規制の失敗」「多様な意見を聞かない文化に問題」「個人の責任が曖昧」といった、言わば事故が無くても指摘可能であったかもしれない分析（あるいは課題）が多く提起された

4.「安全文化の劣化」などの指摘に関し、「その進亢を防ぐ提言/行動を妨げたものは何だったのか」との問いかけに、「殿様体質」「国策民営体制」という原子力運営体制/体質問題を挙げる答えが多かった

Reflections on Fukushima

19th International Conference On Nuclear Engineering (ICONE19) , October 24-25, 2011 Osaka, Japan

Dr. Nils J. Diaz

Managing Director, The ND2 Group, LLC Chairman, ASME Task Force- Japan Events

US Nuclear Power Plants: Rules and Status

- The US is well prepared -overall- to handle severe external events, commensurate with region-wise phenomena.
- U.S. nuclear plant designs are required to consider the most severe of the natural phenomena that have been historically reported for the site and surrounding area with sufficient margin to ensure performance of safety functions, including seismic events, floods, tsunamis, hurricanes, tornadoes.
- U.S. nuclear power plants are designed to cope with a station blackout event that involves a loss of offsite power and onsite emergency power. The NRC's detailed station blackout regulations at 10 CFR 50.63 address this scenario, and now B.5.b supported.
- In the aftermath of the 9/11 terrorist attacks, the NRC moved quickly to ensure that safety margins would be maintained under extreme conditions like life fires and explosions. The so-called **B.5.b requirements,** now codified, do provide extra margin for preventing, minimizing and managing reactor accidents, and were enacted on the basis of adequate protection, even though they were clearly significantly beyond the design basis of reactors.

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Severe Accident Management Considerations – B.5.b –

- The NRC Staff and the nuclear industry developed guidance in 2005 for implementing **B.5.b requirements,** including best practices and strategies for mitigating losses of large areas of the plant and measures to mitigate fuel damage and minimize radiological releases, **using on-site and off-site pre-deployed resources.**
- The B.5.b requirements address low-likelihood, high-consequence events that warranted enhancements to defense-in-depth, redefining the level of protection that was regarded as adequate.
- Practices include adding make-up water to spent fuel pools, spraying water on spent fuel, enhanced initial command and control activities for challenges to core cooling and containment, and enhanced response strategies for challenges to core cooling and containment.
- <u>B.5.b-type safety enhancements, if effectively and timely implemented in Japan, should have mitigated the events facing the operator of the Fukushima Daiichi reactors, and very specifically dealt with "station blackout" and cooling of core and fuel pools.</u>

Root Causes of the Accident

- Japanese nuclear safety people should have been more keen to the prevention of soil contamination due to large releases, after witnessing the tragedy caused by the Chernobyl. The regulators have not promoted the IPEEE and lost the opportunity to identify external events that contribute to large releases of I and Cs and the huge liability cost accompanied with such releases.
- Japanese nuclear regulator and operators, who were shy with probabilistic analysis, failed to let the experts of tsunami know the necessity of having information about a tsunami that has a frequency of exceedance of less than 1 in 10,000 years.
- Before 2000 or so the experts of tsunami had been interested in finding the historical maximum tsunami height at a given site with limited resources available and nuclear people had utilized it as a design basis without paying due attention to the limitation.

Root Causes of the Accident

- They have not vigilantly reviewed the contents of the debate held in the academic circles related with earthquake and tsunami, based on questioning attitude and a commitment to excellence.
- Nuclear regulator and operators have tended to limit their attention to the prevention of accidents within deterministically-set design basis and have failed to satisfy the internationally recognized need for defense-in-depth features that will prevent a disproportionate increase in radiological consequences from an appropriate range of events which are more severe than the design basis event including terrorist attacks (cliff-edge).
- This tendency has come from Japanese nuclear community's shallow understanding of the huge hazard potential of nuclear reactors and the resultant weak attention on the objective to assure the extreme rarity of land contamination.

中央防災会議：防災から減災へ

- 社会の目標：500年、1000年に1回、起こりうる津波を前提にするべき。
  - そんな津波を巨大な防潮堤などの構造物で防ぐのは費用効果比から無理。よって、学校など住民の命に関わる施設は、過去最大規模の津波の浸水域から免れる高台に作るか、容易に避難できる構造にして、避難する場所や経路の整備と訓練などの対策を行うべし。
  - 避難できない原子力発電所はそんな1000年津波にも備えなければならない。
- 深層防護の最後の手段として、避難する仕組みを整備するというのは従来の原子力防災計画整備の考え方でもあるのだが、現在は全く受け入れられていない。社会はこの減災の考え方をどの程度の自然災害から受け入れるのか？