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

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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。

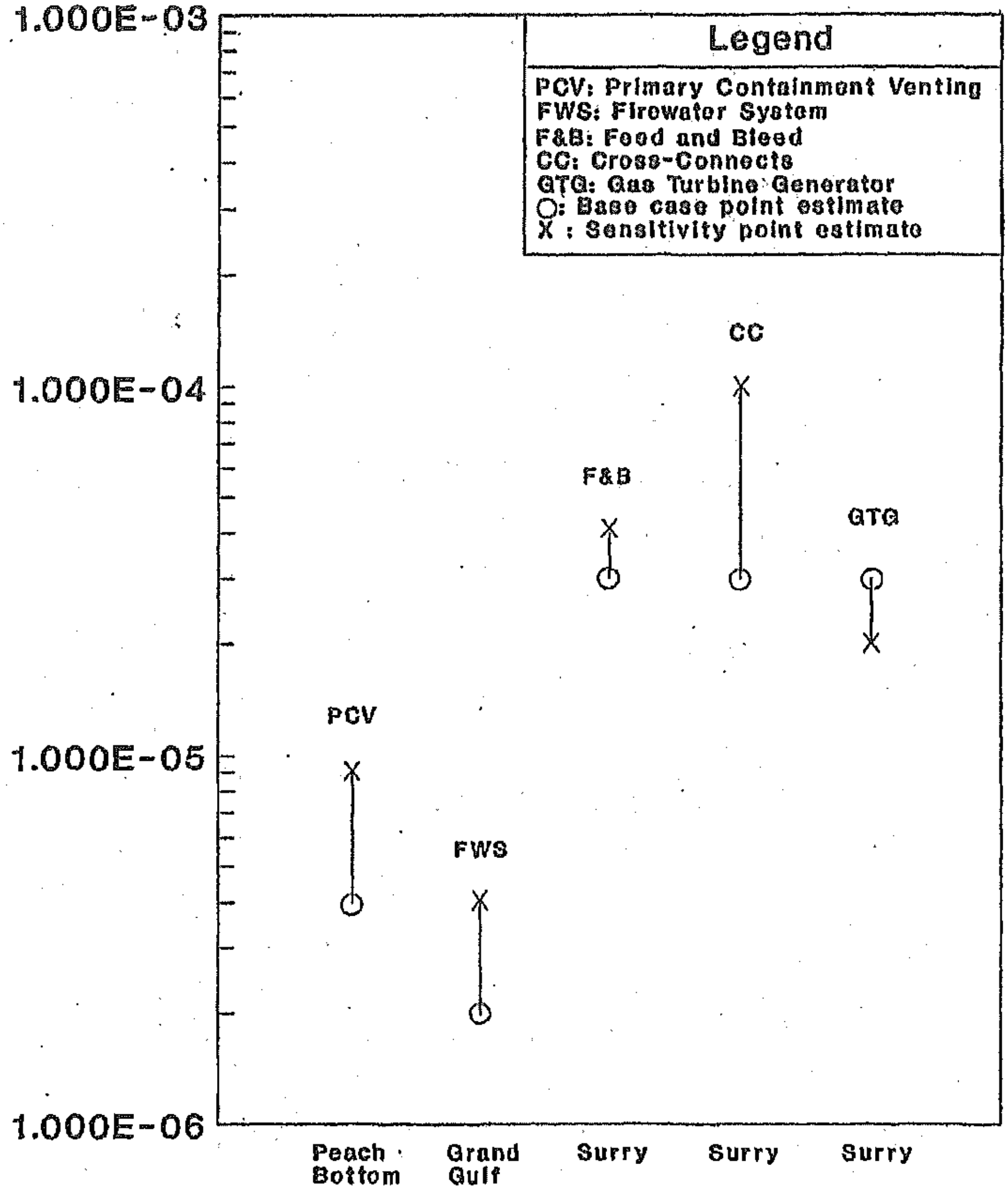


Figure 13.1 Benefits of accident management strategies.

[redacted]

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.

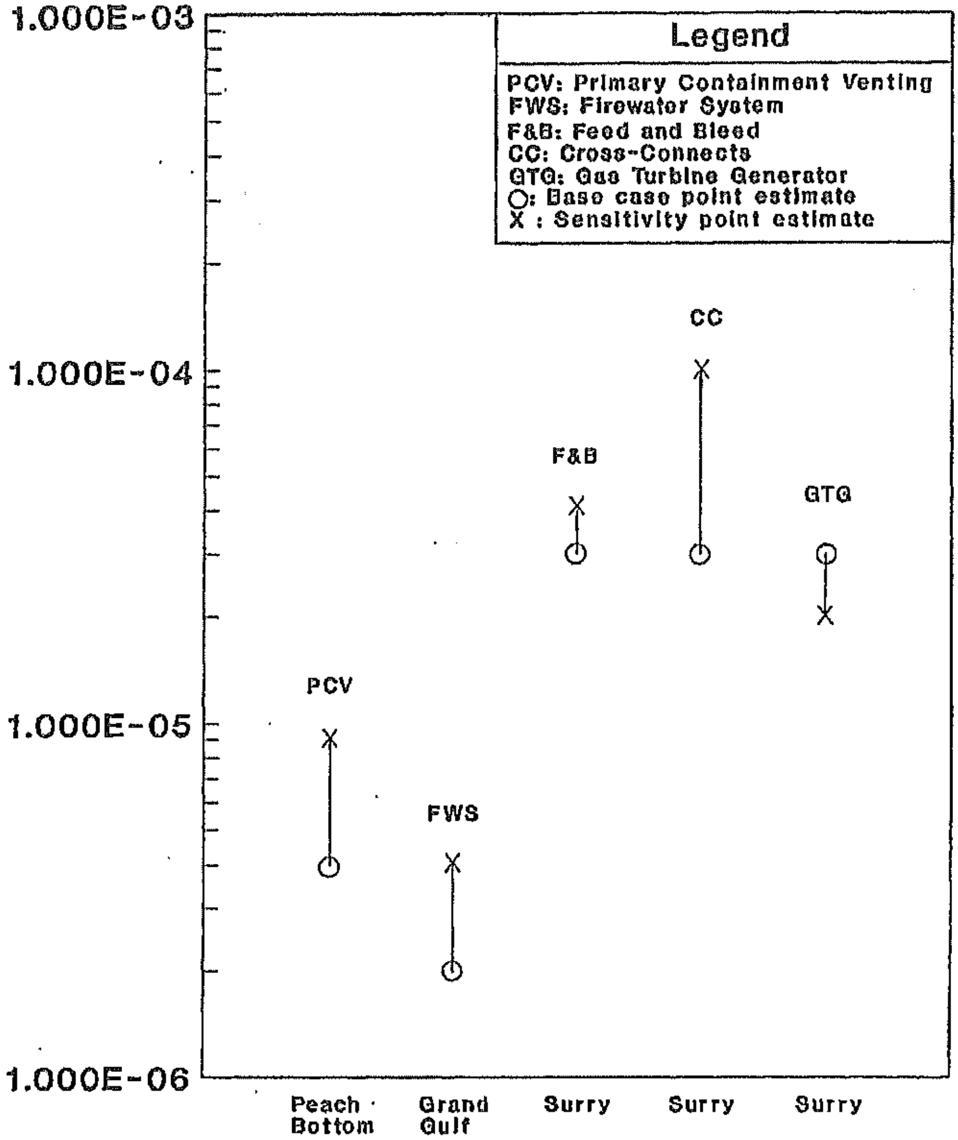


Figure 13.1 Benefits of accident management strategies.

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.

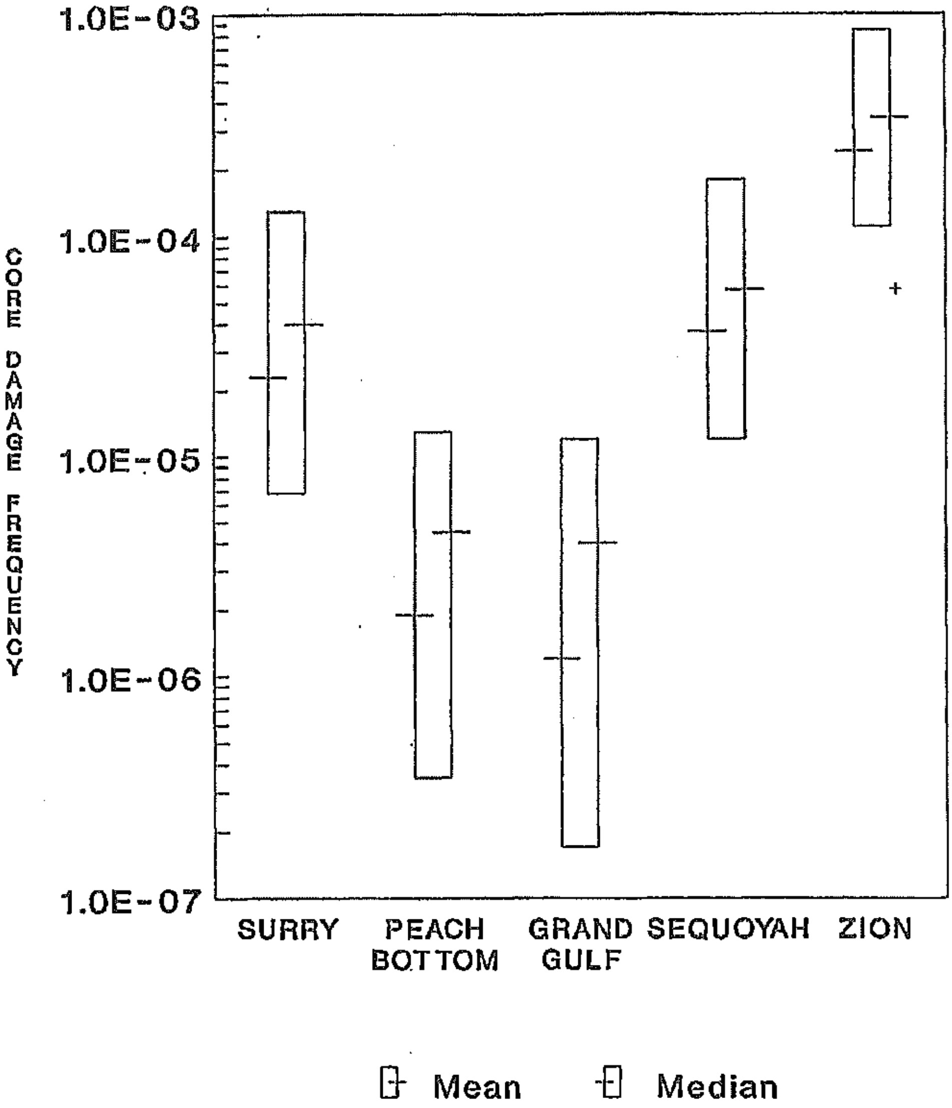
Figures 8.5 through 8.8 provide the results of the external-event analyses, and Figures 8.9 through 8.12 give the breakdown of these analyses according to the principal types of accident sequences.

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).

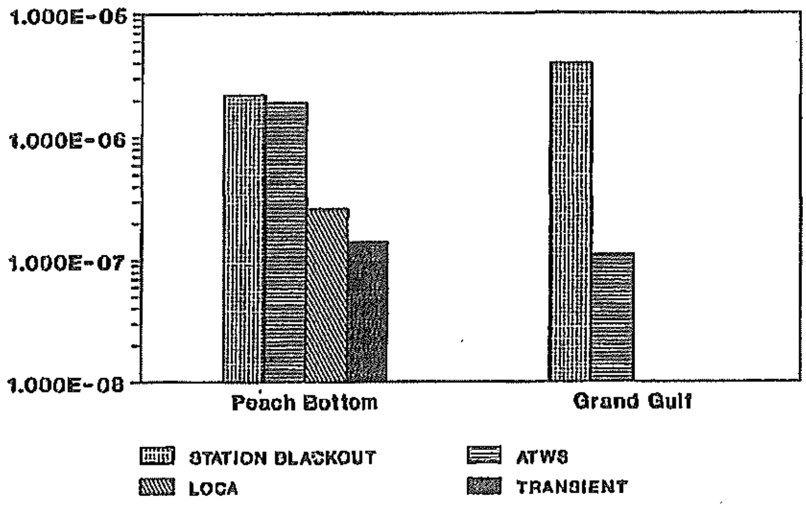
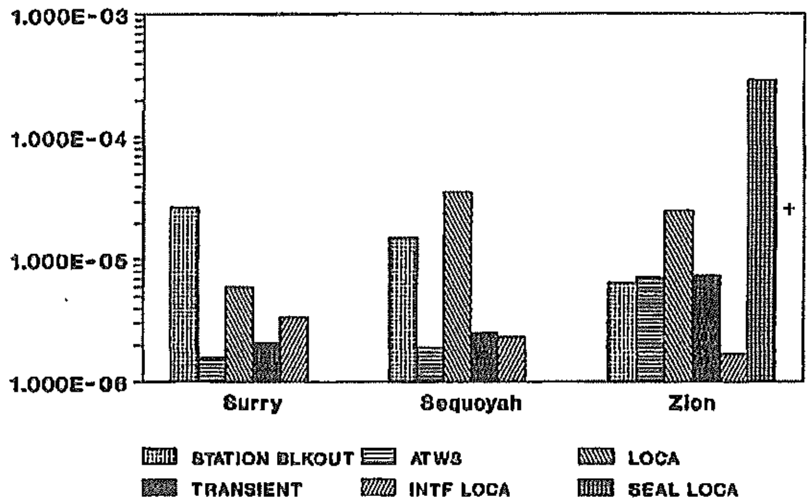


Figure 8.2 BWR principal contributors to internal core damage frequencies.



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.

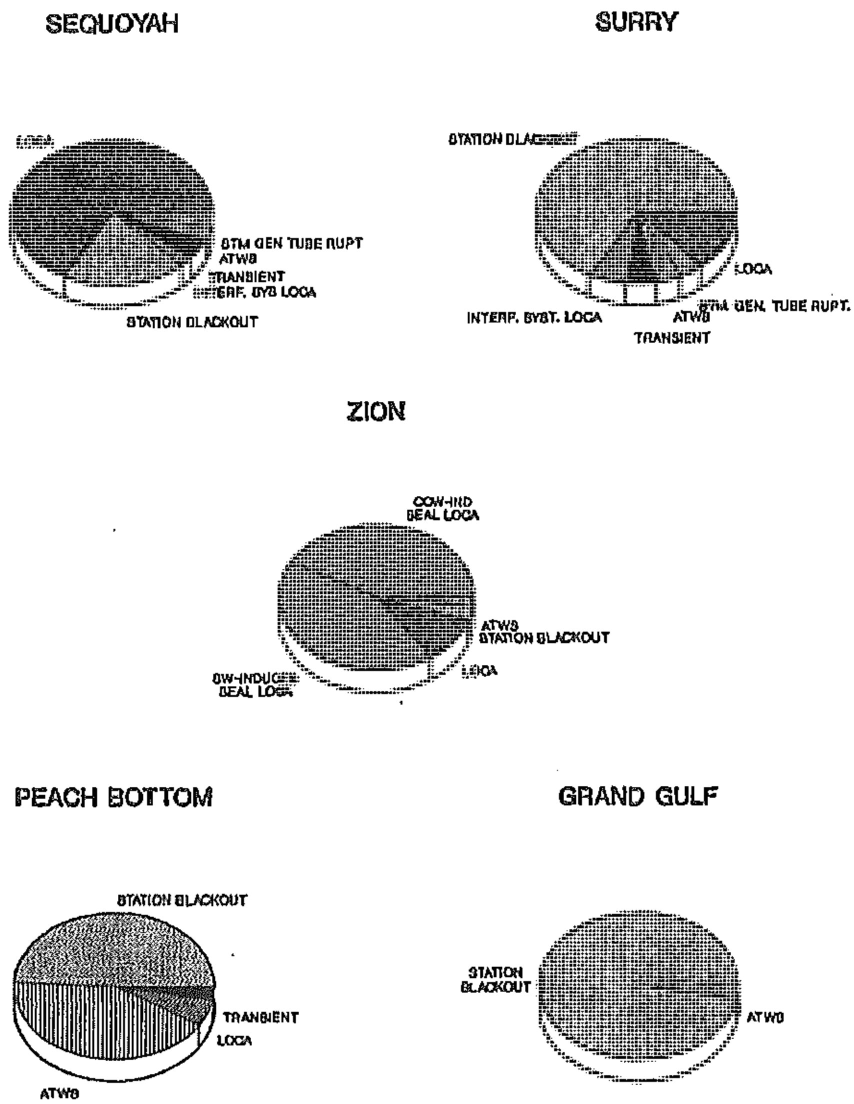


Figure 8.4 Principal contributors to internal core damage frequencies.

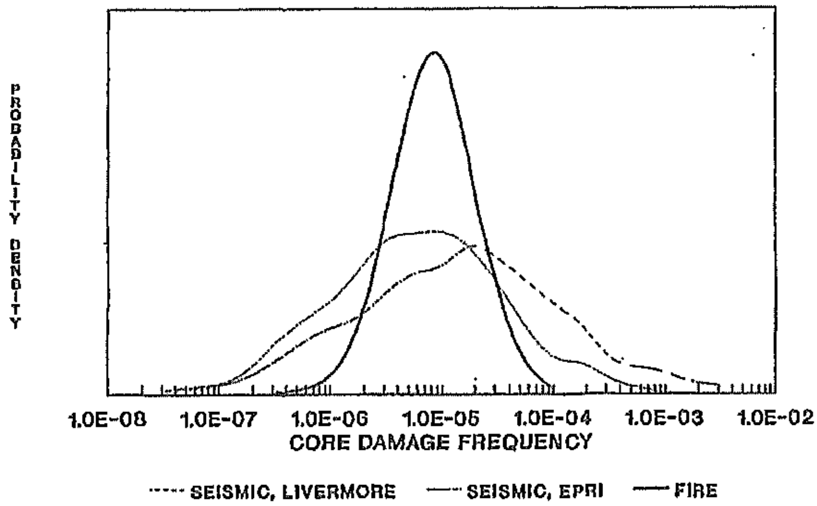
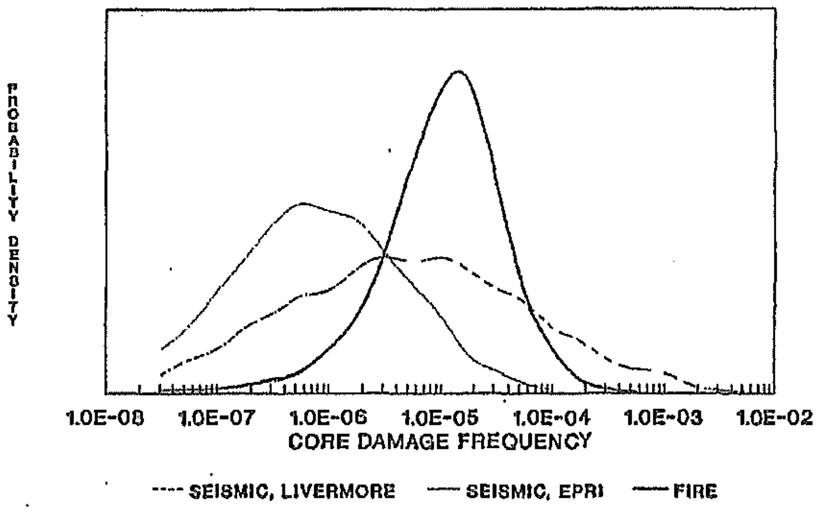


Figure 8.5 *Surry external-event core damage frequency distributions.*



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.

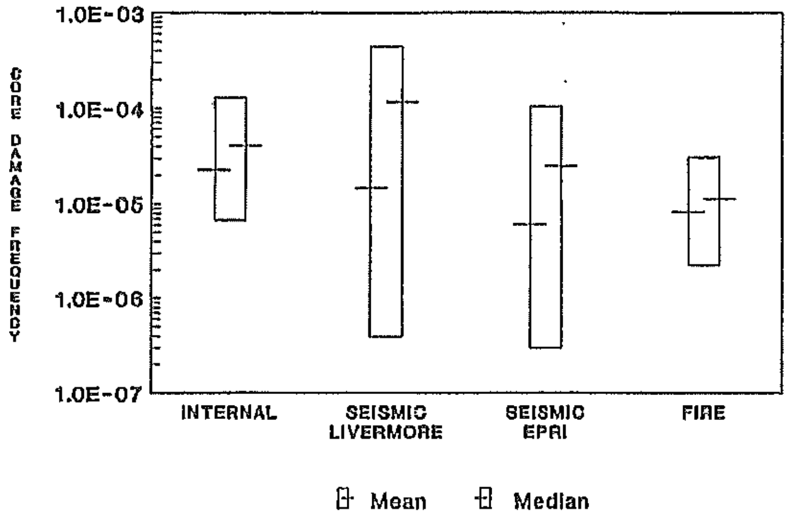
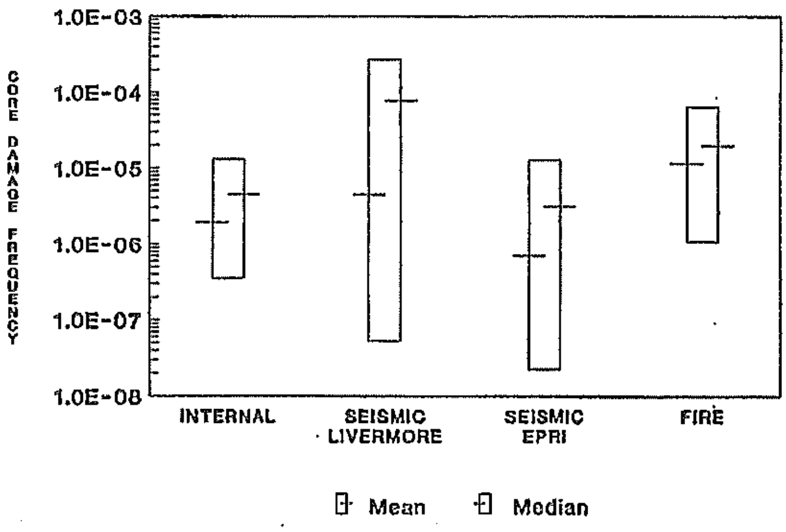


Figure 8.7 Surry internal- and external-event core damage frequency ranges.



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.

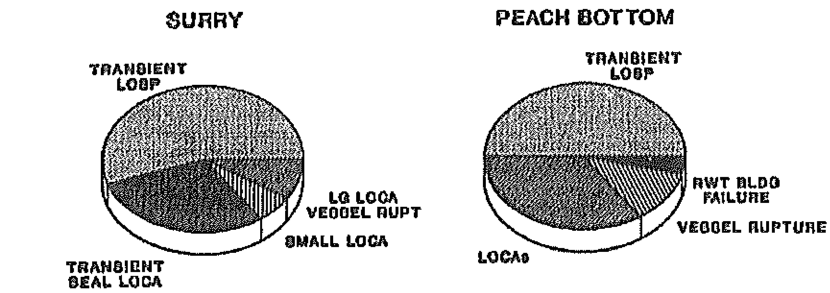


Figure 8.9 Principal contributors to seismic core damage frequencies.

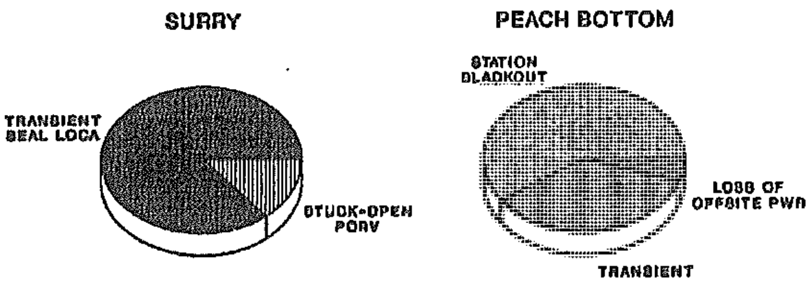


Figure 8.10 Principal contributors to fire core damage frequencies.

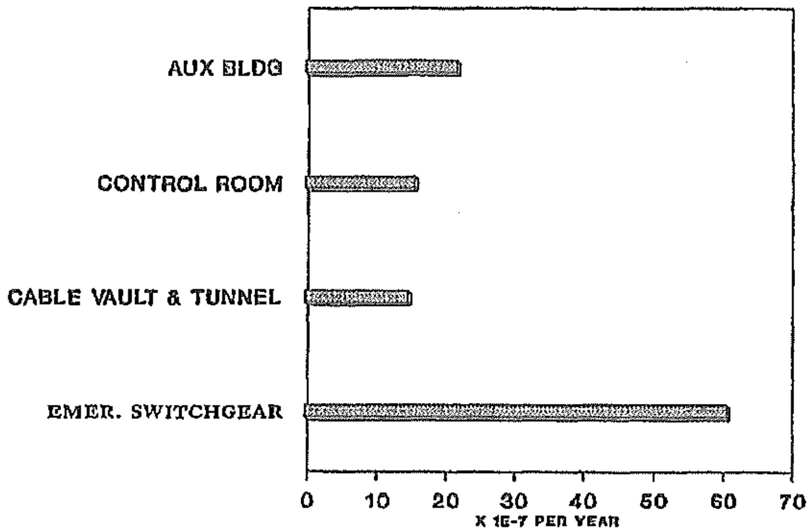
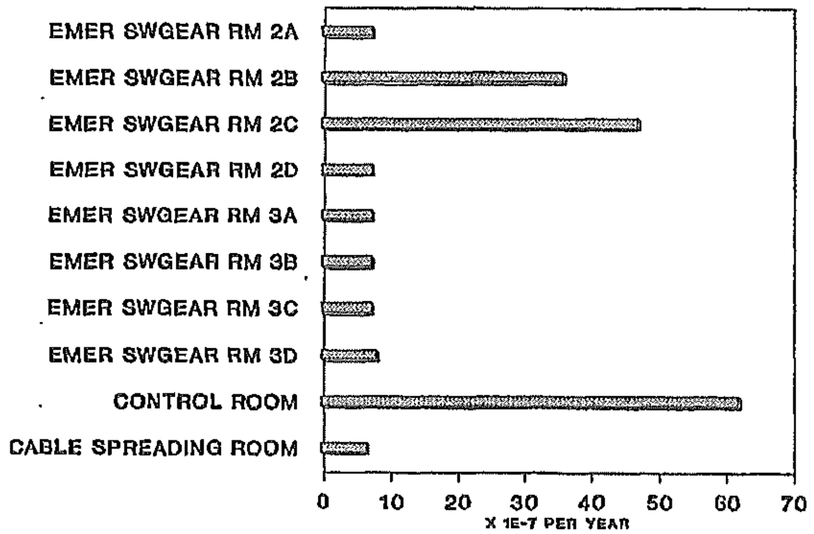


Figure 8.11 Surry mean fire core damage frequency by fire area.



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.

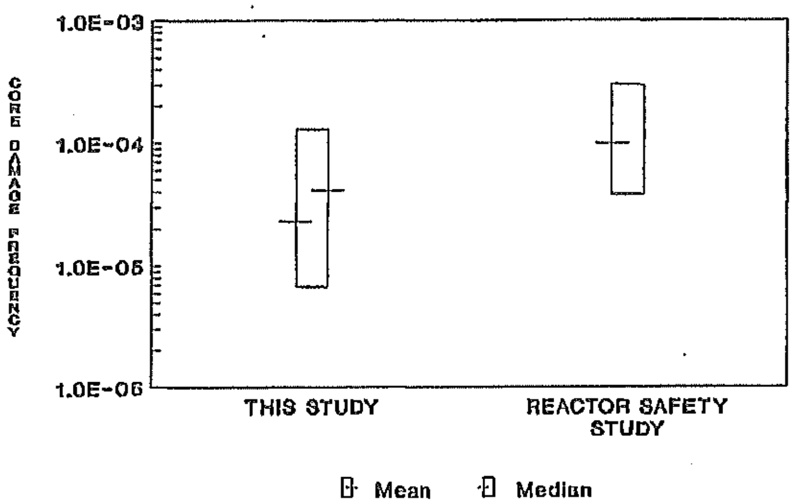
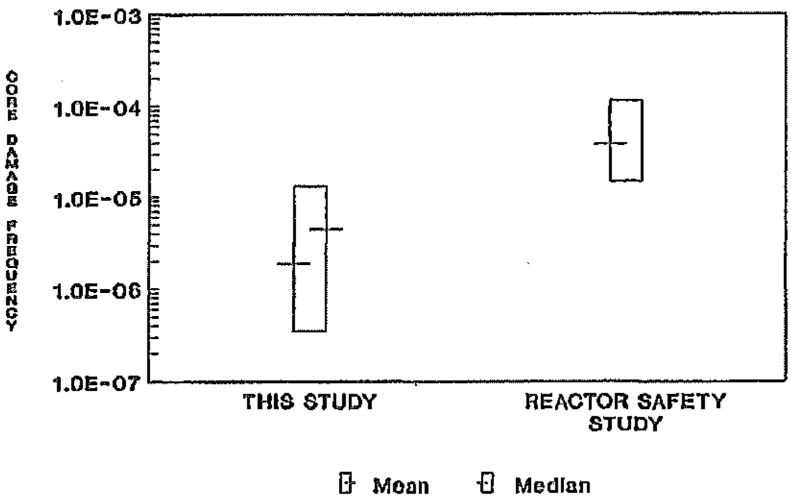


Figure 8.13 Comparison of Surry internal core damage frequency with Reactor Safety Study.



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.