NextEra Energy Seabrook, LLC
(Seabrook Station, Unit 1)
License Renewal Application

NRC Staff Answer to Motion for Summary Disposition of Contention 4B

ATTACHMENT 4B-L

Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice Condenser and Mark III Containment Plants

Final Letter Report

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LIST OF ACRONYMS

AC	alternating current
AFW	auxiliary feedwater
APB	accident progression bin
BEIR	Biological Effects of Ionizing Radiation
BLS	Bureau of Labor Statistics
BNL	Brookhaven National Laboratory
BWR	boiling water reactor
CDF	core damage frequency
CPEF	conditional probability of early (containment) failure
DC	direct current
DCH	direct containment heating
DW	drywell
EF	early failure
FSAR	Final Safety Analysis Report
IPE	Individual Plant Examination
IPEEE	IPE for External Events
LERF	large early release frequency
LF	late failure
LOCA	loss of coolant accident
LOSP	loss of offsite power
MAAP	Modular Accident Analysis Package
NF	no failure
PDS	plant damage state
PRA	probabilistic risk assessment
PWR	pressurized water reactor
RCIC	reactor cooling isolation system
RCP	reactor coolant pump
RCS	reactor coolant system
ry	reactor-year
S	scrubbed
SBO	station blackout
SDP	Significance Determination Process
SNL	Sandia National Laboratories
SP	suppression pool
SPAR	Standardized Plant Analysis Risk
SSW	standby service water
TMI	Three Mile Island
US	unscrubbed

1. BACKGROUND

SECY-00-0198 [1] presented a risk-informed alternative to the current regulation in 10 CFR 50.44 that deals with the threat of combustible gases to the integrity of the containment in light-water reactor nuclear power plants. One of the risk insights developed in SECY-00-0198 indicated that station blackout (SBO) accident sequences represented a threat to containment integrity in boiling water reactor (BWR) plants with a Mark III containment and pressurized water reactor (PWR) plants with an ice condenser containment. These pressure-suppression containments were mandated under 50.44 to install combustible gas igniters that would burn the hydrogen evolved via the metal-water reaction during severe core melt accidents. The igniters are designed to burn the evolved hydrogen at relatively low concentrations and thus reduce the potential for large deflagrations or detonations that could challenge containment integrity. However, the igniters need alternating current (AC) power to operate and would not be available in an SBO accident. Thus, enhancements that would allow combustible gas control during SBO accidents could reduce the risk from combustible gases. The issue to be analyzed is whether such enhancements would be cost beneficial, i.e., whether the averted risk, evaluated in terms of the expected value of averted costs, would be greater than the direct cost of implementation of the enhancement.

Under Project JCN W-6224 Brookhaven National Laboratory (BNL) is providing an estimate of the benefit values associated with making enhancements to the combustible gas control systems in PWR plants with ice condenser containments and in BWR plants with Mark III containments. In addition to calculating benefits based on point estimates or mean values, BNL has also been asked to provide insights into the uncertainty of the estimates provided. This estimate of benefit values is the subject of the present report. The enhancement would make combustible gas control available during SBO accidents, and this could be accomplished in a number of ways. BNL is not considering the implementation costs of any enhancements (these are calculated elsewhere), and therefore, this report is silent on the particular means by which the combustible gas control will be accomplished.

Based on the Statement of Work, this report discusses what averted costs should be included in the analysis and how they should be treated. Avoided (offsite) person-rem and avoided (offsite) property damage are mentioned as potential benefits in the Task Action Plan for Generic Safety Issue 189. The Statement of Work indicates that the analysis should include all types of averted costs in accordance with NUREG/BR-0058, Rev. 3 [2] and the estimation and evaluation of values should comply with Section 4.3 of NUREG/BR-0058, Rev. 3.

2. APPROACH

This report provides an estimate of the benefit accrued from enhancing the currently installed combustible gas control systems in PWR nuclear power plants with ice condenser containments and BWR plants with Mark III containments. The current systems are not available during SBO accident sequences, and the enhancement whose benefit is being estimated would allow combustible gas control during SBO sequences. The analysis presented here is concerned only with the value of the benefit obtained from such an enhanced system, not the details involving what changes, additional systems, etc. are implemented to achieve the enhancements. Note that this means that any negative benefit associated with the installation of the enhancement, such as worker exposure during installation, is not considered here, and is dependent on the particular means chosen to implement the enhancement. It is expected that items such as worker exposure would be included in the estimates for the cost of the enhancement, which is being estimated elsewhere. The benefit calculated here is expressed in terms of the risk averted as a result of the enhancement, stated in terms of current dollars.

The work scope of this project does not allow for a new integrated analysis, but instead calls for estimates based on previously obtained probabilistic risk assessment (PRA) results from a number of different existing studies. This also means that for the evaluation of uncertainty in the estimate no integrated uncertainty analysis is possible. However, some uncertainty information can be obtained from existing PRA models of the relevant plant types.

In terms of current dollars, the averted risk for the enhancement in question, where risk equals likelihood times consequences, is calculated for this study using the following steps:

1. The frequencies of the affected accident sequences are determined in terms of frequency per reactor-year (ry). For the combustible gas control enhancement the applicable sequences are the SBO sequences.
2. The change in conditional containment failure probability for each relevant containment failure mode as a result of the enhancement is determined.
3. The consequences associated with each containment failure mode are determined. If the consequences are in terms of person-rem (such as for health effects) for a population density estimated for a previous year, the person-rem are adjusted by a factor which reflects the estimated change in population density from the year of the calculation to the year 2000. The person-rem are then monetized by a dollar/person-rem factor. If the consequences are in dollars estimated for a previous year (such as for property damage), the dollars are converted to current dollars with an appropriate inflation factor.
4. The product of the conditional containment failure modes times their consequences without the enhancement are summed, as is the product of the conditional containment failure modes times their consequences with the enhancement in place.
5. The sum obtained with the enhancement in step 4 is subtracted from the sum without the enhancement. The difference is multiplied by the frequency determined in Step 1. The result is the averted risk, in terms of dollars per reactor-year.

6. A present value calculation is performed using the result of Step 5, and the remaining years of assumed plant life, to obtain the benefit for the life of the plant in terms of current dollars.

The benefit analysis carried out here are in accordance with the guidance on estimation of values provided in NUREG/BR-0058 [2] and in NUREG/BR-0184 [3]. In particular, in conformance with Section 4.3.2 of NUREG/BR-0058, the estimation of value attributes related to the enhancement considered here include:

- reductions in public and occupational radiation exposure,
- averted offsite property damage, and
- averted onsite impacts

Additional potential value attributes listed in NUREG/BR-0184 are: enhancements to health, safety, or the natural environment; savings to licensees; savings to NRC; savings to State, local, or tribal governments; improved plant availability; promotion of the efficient functioning of the economy; and reductions in safeguards risk. These were not considered in the present analysis because they were deemed to be either not applicable or would have a negligible impact on the results.

In the present analysis, again as called for in NUREG/BR-0058:

- changes in public health and safety from radiation exposure and offsite property impacts are examined over a 50 mile distance from the plant site,
- the recommended dollar conversion factor of $2000 per person-rem is used and used only to capture the health effects attributable to radiological exposure,
- offsite property damage consequences are addressed separately and treated as an added factor in the value assessment,
- estimated values are expressed in monetary terms whenever possible and expressed in constant dollars from the most recent year for which price adjustment data are available,
- all values and impacts are expressed on a present worth basis for lifetime benefits, and
- a discount rate of 7% is used for the present-worth calculation, with a sensitivity analysis at a 3% discount rate.

NUREG/BR-0058 also calls for value estimates to be based on mean or 'expected value' calculations when possible, and to consider uncertainties. However, NUREG/BR-0058 also recognizes that the level of detail available from data sources may not allow expected value estimates to be used, and allows sensitivity analyses, including hypothetical best and worst case values, to be used in lieu of uncertainty analyses. The enhancement under consideration here carries with it no potential reduction in core-damage frequency, only in containment failure probability. The emphasis of the evaluation is on containment performance, i.e., the reduction in the conditional containment failure probability when combustible gas control is available during SBO events. Estimating changes in containment failure probability are especially uncertain and involve sparse data. In addition, the analysis here relies on calculations from previous analyses carried out for other purposes. Therefore, the benefit estimates calculated here are not always based on expected value, and use some sensitivity calculations as well as some previously obtained uncertainty results.

It should also be noted that NUREG/BR-0058 calls for a safety goal evaluation, using certain safety goal screening criteria relative to the enhancement, under some situations. However, as stated at the end of Section 3.3.2 of NUREG/BR-0058, ". . .the safety goal screening criteria described here do not address issues that deal only with containment performance. Consequently, issues that have no impact on core damage frequency (CDF) (delta CDF of zero) cannot be addressed with the safety goal screening criteria." No safety goal evaluation has been carried out in the present analysis.

As noted above, the results presented in this report were calculated based on information gathered from various existing analyses. The severe accident progression scenarios, including conditional containment failure probabilities, are based primarily on the NUREG-1150 [4] work, including the descriptions and values reported in the NUREG-1150 supporting documents for the Sequoyah [5] and the Grand Gulf [6] analyses. The conditional probability of early failure (CPEF) of containment from NUREG/CR-6427, "Assessment of the DCH Issue for Plants with Ice Condenser Containments" [7] was used to for a sensitivity case for the ice condenser estimates. Finally, NUREG/CR-xxxx, "Basis Document for Large Early Release Frequency (LERF) Significance Determination Process (SDP)" [8], which summarizes relevant NUREG-1150 information, was employed to establish the accident progression used for the BWR Mark III estimates. It should be noted that all these references, with the exception of References 7and 8 are included in NUREG/BR-0184, the Regulatory Technical Analysis Handbook, as appropriate references for value impact analysis. References 7 and 8 are too new to be included in NUREG/BR-0184. In addition to the NUREG-1150 SBO frequencies, the frequencies from the Duke Power PRAs contained in Reference 9 were used in the uncertainty considerations. SBO frequencies from the NRC's SPAR models, as well as frequencies reported in the Individual Plant Examinations (IPEs) and in the IPEs for External Events (IPEEEs) for the plants, are also discussed. The value of offsite property damage and offsite person-rem are taken mostly from an earlier BNL study, NUREG/CR-6349 [10]. The exception is some of the values of offsite person-rem for the Duke Power plants, which were extrapolated from Reference 9. Discussion provided on the values of onsite health costs and onsite property damage costs are based on the information provided in Burke, Aldrich and Rasmussen [11], and in NUREG/BR-0184 [3]. Updates of population densities are based on population projections found in the Final Safety Analysis Reports (FSARs) of the plants examined, not on actual current population statistics.

The remainder of this report is organized as follows: Section 3 below provides a discussion of averted costs, i.e., benefits of providing means (such as installing a backup power supply for the hydrogen igniters) to allow combustible gas control to function during SBO accidents. The various categories of applicable costs, including offsite health costs, offsite property damage costs, and the onsite costs, including employee health costs and onsite cleanup and decontamination costs for accidents that fail containment, are discussed and summarized. Sources of data for the various categories of costs are identified and referenced, where relevant.

Section 4 presents the results obtained for a number of PWR ice condenser plants, based on existing studies. An example calculation is provided, along with the results from a number of additional calculations which provide insight into the uncertainties involved in the benefit estimates.

Section 5 presents similar results for BWR Mark III plants, with Grand Gulf as the Mark III surrogate. In addition, some of the Grand Gulf results are extrapolated to another BWR Mark III plant in this Section to obtain a more generic estimate of the benefit that could be obtained for BWR Mark III plants from a combustible gas control system that is operational during station blackout.

In Section 6 the results obtained are discussed, and some reasons for the differences between the PWR ice condenser results and the BWR Mark III results are provided.

3. DISCUSSION OF AVERTED COSTS

The averted costs arise from the averted consequences of reactor accidents. In general, there are several categories of offsite consequences that follow the occurrence of an accident that begins with core melt and progresses to containment failure and the release of radioactive material from the reactor core to the environment: (1) acute effects of large radiation doses generally in excess of 200 rem to offsite populations in the initial phases of the release that can lead to early health effects (early fatalities or early injuries), (2) chronic effects of lower radiation doses that can lead to cancer induction over long periods of time and cause latent cancer fatalities or injuries, and (3) the offsite costs of emergency response and long-term protective actions that are taken to protect the public from radiation.

The risk metrics used to estimate offsite acute and chronic health effects are early (or prompt) fatalities and early injuries and latent cancer fatalities and injuries, respectively. Acute health effects arise soon after exposure via the inhalation, cloudshine, and groundshine pathways. As noted above, acute doses in excess of about 200 rem whole body can lead to early fatality. Chronic effects of long-term exposure are due to three pathways: groundshine from living on contaminated land, inhalation from breathing resuspended radioactive material, and ingestion of contaminated food or water. Dose models embedded in consequence codes predict the dose to a population living in a certain spatial segment based on the characteristics of the release (magnitude, timing, and energy), sampling over the weather at the site, and on any counter measures that are taken. Dose-response models then are used to predict the early fatalities and latent cancers based on the extent of exposure.

The counter measures that are taken to protect the offsite public from the released material involve costs that depend on the nature of the protective measures and their duration. The sum of these costs are usually called the "offsite property damage costs." In the early stages of an accident, costs are associated with emergency evacuation and relocation. These will depend on the number of people affected and the duration of the emergency period. Evacuated individuals will generally remain relocated and will not be allowed to return until the projected groundshine dose is below the protective action guideline value for at least the duration of the emergency phase. In the longer term, people will remain relocated and thus continue to incur costs associated with temporary relocation, depending on the doses from the resuspension inhalation and groundshine pathways. Over a time period of several years following the release, a decision has to be made whether contaminated property, such as farmland and non-farm areas, should be decontaminated or permanently interdicted. The consequence code MACCS, for example, models three successively higher levels of decontamination, each associated with respectively higher costs. If the decontamination efforts plus natural decay cannot reduce the projected long-term dose to an individual below a specified value, or the cost of decontamination exceeds the value of the farmland or non-farm property, then the property or farmland is interdicted and its discounted value is added to the other offsite costs. If people must be permanently resettled because their property is condemned, further costs are added based on estimates of personal income loss and moving costs for a transitional period. Finally, costs are associated with the disposal of contaminated farm products and restrictions on crop, dairy, and meat production from contaminated farmland. Dose criteria associated with protective action guidelines on ingestion of contaminated food are used to determine whether farm products should be discarded.

In value-impact analysis, the averted costs that are ascribed to the averted offsite health impacts are calculated based on the monetary equivalent of averted collective dose (person-rem) at the current NRC-recommended value of $2000 per averted person-rem. They are not calculated based on assigning a monetary value to the early fatality and latent cancer fatality risk metrics. The figure of $2000 per person-rem is assumed to subsume the early and latent fatalities, as well as severe hereditary effects. To obtain the total averted offsite cost (or benefit) of a proposed action, the offsite property damage costs that arise from the long-term protective actions, as discussed above, are added. It should be noted that the costs of long-term protective actions depend on the criteria selected for the allowable dose levels of long-term exposure of the affected population, i.e., there is a trade-off between a higher dose limit/lower cost and a lower dose limit/higher cost. This feature of benefit-cost analysis is discussed at some length in Reference 10.

In addition, there are also potential onsite consequences that are associated with severe accidents. Onsite consequences are not generally modeled in consequence codes, such as MACCS, and NUREG/BR-0058 cautions that particular care should be taken in estimating dollar savings derived from averting onsite costs, since values are often difficult to estimate accurately. There have been a limited number of studies which have attempted to estimate onsite costs. In particular, Strip [12] looked at the impact on worker health, including fatalities and injuries of severe accidents involving core melt and vessel breach. Burke, Aldrich and Rasmussen [11] estimated the cleanup and decontamination costs for both degraded core accidents, such as Three Mile Island Unit 2 (TMI-2), and severe accidents involving vessel breach and possibly containment failure. In the latter case, it is estimated that the cost of cleanup could be significantly higher due to the additional cost of working in high-radiation environments significantly higher than those experienced at TMI-2. A "best estimate" cleanup cost of $1.7 billion (in 1982 dollars) was estimated by Burke and Aldrich for this latter type of accident, compared to half that cost for a TMI-2 type of accident. However, the discussion in Burke and Aldrich implies that the major component of the additional cost is due to the clean-up work carried out in the higher radiation environments due to vessel failure. Since combustible gas control systems cannot reduce the likelihood of vessel breach, but only the likelihood of containment failure, the above difference in cleanup costs does not seem to apply for the case considered in this report. There is no explicit discussion in Burke and Aldrich on the difference between the consequences from accidents that lead to core damage but do not cause containment failure, and those that do involve containment failure.

NUREG/BR-0184, the Regulatory Analysis Technical Evaluation Handbook, does provide some data on occupational exposure that can be used for estimates possibly applicable for the case under consideration here. Section 5.7.3 of this handbook discussed the immediate dose and the long term dose workers may receive during cleanup of a severe accident. For the long term dose three accident scenarios are considered. The difference between Scenarios 2 and 3 appears to be applicable for the case under consideration. Scenario 2 simulates the TMI-2 accident: 50% of the fuel cladding ruptures, some fuel melts, and the containment is extensively contaminated, but there is minimal physical damage. In Scenario 3 all fuel cladding ruptures, there is significant fuel melting and core damage, the containment is contaminated and physically damaged, and the auxiliary building undergoes some contamination. The best estimate long term total exposure for Scenario 2 is 7,640 person-rem, while that for Scenario 3 is 19,760 person-rem. Assuming that the immediate dose is roughly the same for both scenarios, the difference in exposure between the two scenarios is about 12, 000 person-rem. It is not clear from the discussion in NUREG/BR-0184 how much of the additional exposure was due to the containment failure alone, and how much was due to the greater core damage postulated for Scenario 3, and therefore the numbers must be viewed with caution for a situation where the enhancement only addresses containment failure. However, since Scenario 3 explicitly mentions containment failure and the resulting auxiliary building contamination, it would seem that containment failure plays a significant role in the elevated exposure levels of Scenario 3.

It would also seem reasonable to assume that containment failure would have an impact on onsite property damage, since plant equipment and structures outside of containment would be contaminated in such an accident, while remaining relatively uncontaminated if the containment remains intact. Even if the plant is assumed to be unusable after a severe accident with or without containment failure, the net value of the equipment for resale or reuse at another site would be significantly impacted by contamination. Therefore, there would appear to be some benefit from averted onsite property damage when containment failure can be prevented. However, these costs may be small compared to the offsite costs in many cases. But if there is more than one unit at a site, these considerations may be important. For example, Unit 1 at TMI was put back into service subsequent to the accident at Unit 2 after a number of years. Had the TMI-2 containment failed and contaminated the other unit, the start-up of the other unit would most likely have been significantly further delayed or not happened at all. Of course, the Chernobyl accident, where there was no containment, did not prevent the other units on site from restarting eventually, but given the conditions under which these units were restarted, such a restart would have been unlikely in the United States under similar conditions.

The benefit that avoidance of containment failure can have for averting onsite costs associated with a second unit on the same site is difficult to estimate, since it can vary so widely depending on the scenario postulated. For example, replacement power costs, which are the dominant onsite costs, would only occur if it is assumed that contamination resulting from containment failure results in incremental downtime for the accident-free reactor. It is interesting to note that in the case of TMI-2, the accident-free unit remained unavailable for about six years even though it was physically unaffected by the accident at its sister unit. Assuming there was increased unavailability, the magnitude of the replacement power cost would be highly sensitive to when in the reactor's remaining life the accident occurred and the actual number of years of additional unavailability. Given the highly speculative nature and large uncertainties inherent in this type of cost analysis, replacement power considerations will not be included in the total averted cost estimates developed herein.

Among the plant types analyzed in this report, the PWR ice condenser plants are all dual nuclear unit sites (with the exception of Watts Bar, a single unit), while the BWR Mark III plants are all single nuclear unit sites.

Finally, it should be noted that the difference in onsite costs between core melt accidents that involve containment failure, and those that do not, does not appear to have been addressed very well in the literature. A study focusing on this difference could be helpful.

To summarize, the various categories of averted costs that are used in the analysis presented below include:

(1) Offsite Health Costs: These are based on the 50-mile radius offsite population dose (person-rem) associated with the release, conditional on the failure mode, and monetized at $2000/person-rem.

(2) Offsite Property Damage Costs: These are primarily based on the 50-mile offsite costs reported in Reference 10. The 1990 costs shown in Reference 10 have been updated to 2002 dollars using the inflation calculator provided on the Bureau of Labor Statistics (BLS) website [13].

(3) Onsite Employee Health Costs: A value of 20,000 person-rem is used here for occupational exposure for severe accidents with containment failure. A value of 8,000 person-rem is used for occupational exposure for severe accidents without containment failure. These values are based on the results found in NUREG/BR-0184 and discussed above. The person-rem are monetized at $2000/person-rem.

The present worth calculation, i.e., the discounted value of the benefit of the enhancement over the remaining lifetime of the plant (assumed to be 40 years for the plants considered, taking a life extension of 20 years into account) is calculated using the expression $\int \exp(-rt)dt$, where r is the discount rate. Calculations have been performed for the base case of r = 7% and the alternative sensitivity case of r = 3% as recommended in Section 4.3 of NUREG/BR-0058.

4. RESULTS FOR A PWR ICE CONDENSER PLANT

To carry out an estimate of averted costs in accordance with NUREG/BR-0058 and NUREG/BR-0184, risk results in terms of offsite and onsite person-rem, as well as costs, are desired. This means the results from a Level 3 PRA are needed. The NUREG-1150 study for Sequoyah was an integrated study (Level 1, Level 2, and Level 3) PRA study of an ice condenser plant, and there is a significant amount of information regarding accident progression and hydrogen combustion available for Sequoyah as a result of the NUREG-1150 studies. The NUREG-1150 Sequoyah study also provides, separately, uncertainty ranges for CDF (Level 1) as well as containment failure probability (Level 2). However, only internal events were examined for Sequoyah in the NUREG-1150 study. Sequoyah core damage frequency ranges due to SBO events are presented in Table 5.2 of NUREG-1150 Volume 1 [4]. A histogram of the conditional probability of early failure (CPEF) of containment, conditional on loss of offsite power (LOSP) for Sequoyah, is shown in Figure 2.5-2 of NUREG/CR-4551, Vol.5, Rev 1, Part 1 [5]. Table 2 below summarizes the values in the reports.

Table 1 Sequoyah Uncertainty Ranges for Internal Events

	5th	Mean	95th
SBO CDF frequency from NUREG-1150 (ry)	5.2E-7	1.5E-5	5.3E-5
CPEF due to LOSP from NUREG/CR-4551, Vol.5	1.3E-4	0.15	0.65

The percentile frequencies from long and short term SBO have been added to approximate a total SBO percentile frequency in the above table.

4.1 PWR Ice Condenser Example Benefit Calculation

The benefit calculation for Sequoyah using mean values is carried out below, following the steps found at the beginning of Section 2 of this report:

Step 1 - Frequencies of SBO sequences

As indicated in Table 1, the mean SBO core damage frequency from the NUREG-1150 study for Sequoyah is 1.5E-5 per reactor-year (ry) from internal events.

Step 2 - Change in conditional containment failure probability

As shown in Table 1, the mean conditional early containment failure probability due to hydrogen combustion events during SBO in Sequoyah, based on the results of NUREG-1150, is 0.15.

Some credit for random ignition of pre-existing hydrogen was taken in this analysis. The benefit calculations carried out in the present report assume that the enhanced combustible gas control system will be fully effective in reducing the early failure probability to zero. There is a possibility that even if early failure is averted, the accident could proceed to late failure from over-pressurization late in the accident sequence due to steam and non-condensible gases. The presence of functional combustible gas control is not likely to make much difference to the conditional probability of late failure. However, recovery of AC power late in the accident, assuming early failure is prevented, could lead to other systems becoming functional that would allow containment to remain intact. Hence, two possibilities are analyzed: (1) there is no late failure and containment remains intact if early failure is prevented, and (2) late failure occurs even if early failure is prevented.

The pertinent conditional containment failure probability cases are summarized in Table 2.

Table 2 Conditional Containment Failure Probabilities for Sequoyah

Gas Control	Late Failure	CPEF	CPLF	CPNF
no	no	0.15	0	0.85
yes	no	0	0	1.0
no	yes	0.15	0.85	0
yes	yes	0	1.0	0

Where: CPEF is conditional probability of early failure
CPLF is conditional probability of late failure
CPNF is conditional probability of no failure

Step 3 - Consequences associated with each containment failure mode

Offsite consequences for releases representative of both early and late containment failure are presented in Table 3 below for Sequoyah. Offsite person-rem and the offsite property cost estimates are based on the data provided in References 10 for Sequoyah. These results are conditional consequences (i.e., conditional on occurrence of the release) out to 50 miles from the plant and include offsite population dose (person-rem) and offsite damage costs. The release categories for Sequoyah, i.e., source terms, are based on the results presented in the NUREG-1150 study. It is assumed that there are zero offsite consequences associated with no containment failure.

Two values for offsite person-rem are shown for Sequoyah. The 1990 values are based on Reference 10. The 2000 values have been updated based on the change in population density from 1990 to 2000 as estimated in the Sequoyah FSAR. The change is an increase of about 9%. Two values are also shown for the offsite property damage costs. The first is taken from Reference 10 and is in 1990 dollars. The second updates the 1990 dollar values to current year dollars based on the price inflation calculator (approximately 36% over the 1990-2002 period) of the U.S. Bureau of Labor Statistics (www.bls.gov).

Table 3: Offsite Consequences (50-mile radius) of Containment Failure Releases at Sequoyah

Failure Mode	Offsite Person-rem 1990	Offsite Person-rem 2000	Offsite Health Effects ($k)	Offsite Property 1990 ($k)	Offsite Property 2002 ($k)
Early	2.8E+06	3.1E+06	6,100,000	4,800,000	6,600,000
Late	5.2E+05	5.7E+05	1,100,000	500,000	680,000

The sequence used for Sequoyah for early failure consequences is SEQ-11-2 from Reference 5 which is also used in Reference 10. This is a typical early failure sequence with about 88% of noble gases, 29% of iodine, 26% of cesium, and 21% of tellurium released. The late failure sequence used for Sequoyah is SEQ-06-1 from Reference 5 and Reference 10. This is a typical late failure sequence with all noble gases, about 8% of iodine, 1% of cesium, and less than 1% of tellurium released. The discussion in Reference 5 indicates that in both these sequences the ice bed was functional and had some mitigating effect on the releases. It should be noted that the (1990) consequences reported in Reference 10 differ somewhat from those reported in the NUREG-1150 reports, even though Reference 10 is based on the NUREG-1150 analyses. This is primarily because in the NUREG-1150 study the consequence analysis was carried out using Version 1.5.11 of the MACCS code, while the consequences in Reference 10 were recalculated with Version 1.5.11.1 of MACCS. This later version explicitly incorporates the higher Biological Effects of Ionizing Radiation (BEIR) V risk coefficient for the latent cancer-dose relationship while the earlier version of MACCS used the BEIR III risk coefficient. In addition, a few input errors in the NUREG-1150 MACCS calculations were corrected for the recalculations of Reference 10.

Onsite health consequences are calculated assuming 20,000 person-rem occupational exposure, or $40,000k after using the $2000/person-rem factor, for both early and late containment failures, and 8,000 person-rem, or $16,000k, for no containment failure. Onsite property damage is not included as per the discussion in Section 3.

Step 4 - Summation of conditional containment failure modes and their consequences

The results of the summation of conditional containment failure modes and their consequences for the cases outlined above are shown in Table 4.

Table 4: Summation of Offsite Costs and Onsite Health Effect Costs			
Gas Control	Late Failure	Total Offsite Cost ($k) conditional on SBO	Onsite Health Effects Cost ($k) conditional on SBO
no	no	1,900,000	20,000
yes	no	0	16,000
no	yes	3,400,000	40,000
yes	yes	1,800,000	40,000

Step 5 - Subtraction of costs and multiplication by frequency

The calculation in Step 4 was made with and without the gas control system present. The control system is assumed to be fully effective in preventing early failure. The difference between the cases where gas control is 'yes' and the cases where gas control is 'no,' when multiplied by the SBO frequency, represents the averted offsite cost on a per reactor-year basis.

The results are summarized for Sequoyah for accidents with and without late failure in Table 5 below. Costs are divided into offsite and onsite costs, as well as total costs. Offsite costs are the dominant contributor in all cases. Costs are in 2002 dollars.

Table 5: Sequoyah Cost Summary per reactor-year				
Internal Events	SBO Frequency	Total Averted Offsite Costs $k per reactor-year	Averted Onsite Health Effects Costs $k per reactor-year	Total Averted Costs $k per reactor-year
No Late Failure	1.5E-5	28	0.053	28
with Late Failure	1.5E-5	24	0	24

Step 6 - Calculation of lifetime benefit

Multiplication by the present worth factor, based on the discount rate selected and plant lifetime remaining, yields the total averted offsite cost, or benefit, over the plant's lifetime. Results for a lifetime of 40 years for a discount rate of 7% and 3% are shown in Tables 6 and 7 respectively. This step completes the analysis.

Table 6: Lifetime Benefit Base Case (7% Discount Rate)			
Internal Events	Lifetime Averted Offsite Costs 2002$k	Lifetime Averted Onsite Health Effects Costs 2002$k	Lifetime Total Costs Averted 2002$k
No Late Failure	370	0.7	370
with Late Failure	320	0.	320

Table 7: Lifetime Benefit Sensitivity Case (3% Discount Rate)			
Internal Events	Lifetime Averted Offsite Costs 2002$k	Lifetime Averted Onsite Health Effects Costs 2002$k	Lifetime Total Costs Averted 2002$k
No Late Failure	650	1.2	650
with Late Failure	560	0.	560

The results are dominated by the offsite costs. Inclusion of averted onsite costs produces a negligible change in all cases. However, since the ice condenser containments are mostly dual units, the discussion of Section 3 regarding onsite costs related to the effect of containment failure of the damaged unit on the undamaged unit may apply. This means that for the case where containment failure is averted, the onsite averted costs could be significantly higher than estimated here, under certain conditions, as discussed in Section 3. However, if late containment failure occurs, the benefit from averted onsite costs is likely to be very small. This is due to the assumption that the main driver is the additional cost of site cleanup and decontamination of the undamaged unit from failure of containment of the damaged unit. This cost is assumed to be the same whether containment fails early or late. Thus, combustible gas control will offer very little benefit in terms of onsite costs if late failure occurs.

4.2 PWR Ice Condenser Uncertainty Considerations

When considering uncertainties in the results, uncertainties in the Level 1, Level 2, and Level 3 analyses should be accounted for.

For the issue of combustible gas control in containment this means that the uncertainties to be considered are:

1. the uncertainty in the CDF contribution from SBO.
2. the uncertainty in the CPEF due to gas combustion, given SBO has occurred, and
3. the uncertainty in the releases and associated consequences.

In practice to date, a number of studies have provided estimates of (1). Very few have included (2) and/or (3).

To estimate the uncertainty in benefits achieved by enhancing gas control in ice condenser containments to operate under SBO conditions, BNL:

1. made additional benefit estimates based on the uncertainty results from the NUREG 1150 study for Sequoyah [4,5],
2. reviewed some PRA results recently provided by Duke Power from their PRAs of the Catawba and McGuire plants [9] and calculated benefits with the results provided in these models,
3. ran the latest available SPAR models for Catawba and McGuire and calculated benefits based on the uncertainty in the SBO frequencies provided in these models, and
4. reviewed the IPEs and IPEEEs for variation in SBO CDF and variation in CPEF for ice condensers.

NUREG-1150 Sequoyah uncertainty results

Table 1 above summarized the 5^{th} percentile, mean, and 95^{th} percentile values for both the SBO CDF frequency and the CPEF found for Sequoyah in the NUREG-1150 study.

Unfortunately the NUREG-1150 reports do not present the integrated uncertainty from the SBO CDF distribution convolved with the CPEF distribution. However, Figure 2.5-5 of NUREG/CR-4551, Vol.5, Rev 1, Part 1 [5] provides some insight on the range of the combined uncertainties. That figure, which presents frequency distributions of various accident progression bin (APBs) groups, indicates that the 95^{th} percentile of the frequency (i.e., the CDF combined with conditional failure probability) of various scenarios involving early containment failure is no more than one order of magnitude larger than the mean value of the frequency. This data can be used to estimate an upper bound of the 95^{th} percentile of the combined uncertainty by arguing, based on the Figure 2.5-5 results, that the additional uncertainty introduced by the CPEF variability will be limited to an increase of 10 times the result obtained with the CDF and CPEF mean value. This is less than a value obtained by using the 95^{th} percentile SBO CDF and the 95^{th} percentile CPEF to calculate benefit, which would obviously represent a more extreme value than the 95^{th} percentile of the combined uncertainty distribution.

For a lower bound, Figure 2.5-5 is not much help since the 5^{th} percentiles of the frequencies in that figure are more than 3 orders of magnitude below the mean. However, a lower bound on the benefits from the SBO distribution alone results in very low values (as shown below in Table 12), so a combined lower bound is not of interest.

The benefits for Sequoyah were also calculated using the conditional early containment failure probabilities due to hydrogen combustion events during SBO based on the results of NUREG/CR-

6427 [7]. This is a recent, detailed study of severe accident phenomena in ice condenser containment plants, focused on the direct containment heating issue, carried out by Sandia National Laboratories (SNL), which assigns a very high CPEF due to hydrogen for Sequoyah.

PRA results recently provided by Duke Power

In an email communication of September 20, 2002 Duke Power provided selected results from their latest PRAs for the Catawba and McGuire plants. These results consisted of:

1. SBO CDFs for internal events (but including tornado), with point estimates, mean, median, 5th and 95th percentiles of CDF provided, (Three different cases were provided for Catawba), and point estimates of selected SBO CDF's for external events (tornado and seismic).
2. ranges of containment failure probabilities associated with the relevant SBO plant damage states used in the PRA,
3. early containment failure public health risk results, including person-rem per year, from the studies, and
4. definitions of the early failure release classes used to obtain the health effects.

The relevant core damage frequencies provided by Duke are shown in Table 8 below:

Table 8 SBO Core Damage Frequencies (per ry)

Plant Conditional Containment Failure Probabilities	**Internal Events**				**External Events**
	Pt Est	**5th**	**mean**	**95th**	**Pt Est**
Catawba	*Duke PRA Rev 2b*				
Prob of early failure range: 0.16 to 0.21 - slow SBO 0.16 to 0.34 - fast SBO Prob of late failure range: 0.72 to 0.84 - slow SBO 0.68 to 0.84 - fast SBO	1.5E-5	9.4E-7*	1.9E-5*	6.4E-5*	1.0E-5
	Duke Rev 2b with RCP seal replaced				
	9.8E-6	5.2E-7*	1.3E-5*	4.5E-5*	NA
	Duke Rev 2b w RCP seal replaced & flood wall installed				
	1.2E-6	1.5E-7*	2.6E-6*	8.7E-6*	NA
McGuire	*Duke PRA Rev 3*				
Prob of early failure range: 0.15 to 0.19 - slow SBO 0.16 to 0.26 - fast SBO Prob of late failure range: 0.34 to 0.56 - slow SBO 0.17 to 0.36 - fast SBO	1.2E-6	2.2E-7*	3.0E-6*	9.9E-6*	8.9E-6
* includes SBO frequency due to tornado					

With regard to item (3), it was noted that person-rem results for early failures seemed less by a factor between 3 and 4 than those found for NUREG-1150 early failures from comparable scenarios. This difference in health risk was then traced to differences between item (4) above and the release classes from NUREG-1150 for comparable scenarios. Table 9 below shows the differences between a typical release class from item (4) and a typical NUREG-1150 release.

Table 9

	Catawba values based on email from Duke	Sequoyah values based on NUREG-1150
Release Fractions		
Xe	1.0E+00	8.8E-01
I	5.5E-02	2.9E-01
Cs-Rb	4.8E-02	2.6E-01
Te-Sb	3.0E-02	2.1E-01
Ba	1.7E-03	6.5E-02
Ru	2.2E-03	6.0E-03
La	1.2E-04	8.0E-03
Sr	2.5E-04	6.4E-02

As can be seen from this table, the NUREG-1150 release fractions for the important radionuclides are about a factor of 4 higher than the ones used in the Duke PRA. The Duke results were obtained using the Modular Accident Analysis Package (MAAP) code, while the NUREG-1150 results were obtained with the Source Term Code Package and MELCOR. Apparently the differences in the release fractions in the above table are primarily attributable to the use of the different codes in the two analyses.

SPAR Model Runs

BNL ran the latest available SPAR model for the Catawba and McGuire plants, i.e., the 3i model, and calculated benefits with results from these models. These are internal events, Level 1 models which incorporate uncertainty parameters and can calculate, in addition to a point estimate, the mean, median and 5th and 95th percentiles associated with the CDF of a particular accident class, such as SBO. The SBO frequencies used in these models are listed in Table 10 below:

Table 10 SPAR 3i SBO CDF ranges for internal events (ry)

Plant	**5th**	**Mean**	**Point Estimate**	**95th**
Catawba	6.8E-7	2.4E-5	2.8E-5	9.6E-5
McGuire	1.6E-6	2.4E-5	2.2E-5	8.5E-5

Since these are Level 1 models only, there is no information on CPEF or accident progression in the models, and the benefit analyses had to use the Sequoyah NUREG-1150 accident progression and source terms. Therefore, benefit results are directly proportional to the ratio of the SBO frequencies shown in Table 10 and those for Sequoyah shown in Table 1. As can be seen, the SPAR model frequencies for Catawba and McGuire are somewhat (30% to 80%) higher than the NUREG-1150 Sequoyah sequences. The SPAR model frequencies are also significantly higher than the SBO frequencies for Catawba and McGuire in the Duke Power PRAs, discussed above.

However, these models have not undergone a quality assurance process as yet, and the model software warns the user that the 3i versions are developmental versions that have not been peer reviewed, may contain errors, and may change. After receipt of the Duke Power results for the Catawba and McGuire plants, which are based on more up to date information, it was decided not to include the SPAR model benefit results for Catawba and McGuire in this report.

IPE and IPEEE Comparisons

The PRAs conducted for the IPE Program and the IPEEE Program did not include uncertainty estimates. However, a survey of the SBO frequencies and containment failure probabilities used in the IPEs and IPEEEs was carried out for this report and the results are shown in Table 11, including some of the reasons for the variation in frequency.

Table 11 SBO Frequencies from the IPEs (ry)

Plant	Internal Events	External Events	Additional information from IPEs
Catawba	1.5E-5	1.4E-5	SBO mainly from internal floods. Without floods frequency<10E-6. Shares diesel generator from safe shutdown facility. Low probability for failure to restore offsite power.
D.C. Cook	1.2E-6	5.3E-6	IPE states offsite power very reliable. Auxiliary Feedwater (AFW) manually controlled after battery depletion.
McGuire	9.3E-6	2.3E-5	Standby shutdown facility can provide seal cooling.
Sequoyah	5.3E-6	not available	Can cross-tie DC to operate turbine driven AFW.
Watts Bar	1.7E-5	not available	Short term SBO is an important contributor.

As Table 11 indicates, the internal events SBO CDFs for ice condenser plants in the IPEs are in the range of, or below, the Sequoyah NUREG-1150 mean SBO frequency used in the benefit calculations in this report. The external event frequencies for Catawba and McGuire in the IPEs are considerably higher than the frequencies listed in the current Duke Power PRAs for these plants, as shown in Table 8.

The total (conditional on core damage, not just on SBO) CPEFs in the IPEs for the ice condenser plants were all surprisingly low, i.e., ~0.02 or less, and even smaller than CPEFs for large dry containments. Therefore, benefit calculations based on the IPEs for ice condenser plants would yield significantly lower dollar values than the benefits calculated with the Sequoyah NUREG-1150 numbers or the Catawba and McGuire Duke Power input.

Variation in population density around the plant sites was also surveyed. Based on FSAR projections, McGuire has the highest projected year 2000 (50 mile radius) population density, about 2.3 times that of Sequoyah, which has the lowest. The Catawba population is projected as 1.8 times that of Sequoyah, D. C. Cook's is 1.3 times, and Watts Bar's is about the same as Sequoyah.

4.3 Summary of PWR Ice Condenser Results

Table 12 summarizes the results of the calculations carried out for estimating the benefit of an enhanced combustible gas control system for the ice condenser plants. Results, in terms of averted costs in $k, are shown for 3 Sequoyah cases, 9 Catawba cases and 3 McGuire cases. The columns in the table are arranged as follows:

Column 1 provides the plant name and the case number.

Column 2 lists the containment failure probabilities used and their source.
N1150 refers to the NUREG-1150 study and the supporting documents [4,5,6].
N/C 6427 refers to the SNL report NUREG/CR-6427 [7].
Duke PRA range refers to the ranges provided in the Duke email of 9/20/02 [9].

Column 3 indicates the source used to calculate the consequences.
1150S refers to the NUREG-1150 parameters for Sequoyah, but updated to the values used in NUREG/CR-6349 [10].
Duke refers to the parameters used in the Duke PRA [9].
1150S*1.8 and 1150s*2.3 refers to the 1150S values scaled by a factor for differences in population density.

Columns 4 - 7 give averted costs in $k for internal events obtained by combining
the SBO frequencies obtained from a point estimate (col 4), the 5th percentile (col 5), the mean (col 6), and the 95th percentile (col 7), each combined with the containment failure probabilities shown in column 2.

Column 8 gives the internal events averted cost estimate approximating the upper
bound 95th percentile of the combined SBO CDF and CPEF uncertainty, based on the discussion of Figure 2.5-5 of NUREG/CR-4551, Vol.5, Rev 1, Part 1, provided above.

Column 9 provides the averted cost based on the external events SBO frequency, for
which only point estimates exist.

The PRA source of the SBO frequencies for each plant are indicated across the columns.

Table 12 Averted Costs ($k)

Plant	Case: Cond Cntmt Failure Prob	Case: Source Term	Internal Events: Point Estimate	Internal Events: Uncertainty 5th	Internal Events: Uncertainty mean	Internal Events: Uncertainty 95th	Internal Events: Upper Bound Estimate of 95th Combined (Lv1 & Lv2) Uncertainty	External Events: Point Estimate
Sequoyah			*NUREG-1150*					
1	EF =0.15 (N1150 mn)	1150S (updated)	NA	11	320	1,200	3,200	NA
2	EF =0.65 (N1150 95th)			50	1,400	5,000		
3	EF=0.97 (N/C 6427)			74	2,100	7,500		
Catawba			*Duke PRA Rev 2b*					
1	EF=0.29 LF=0.71 (N/C6427 & Duke PRA range)	Duke	180	11*	220*	750*	2,200*	120
2		1150S	640	40*	790*	2,700*		420
3		1150S*1.8	870	54*	1,100*	3,700*		580
			Duke Rev 2b with reactor coolant pump (RCP) seal replaced					
4	same as above	Duke	120	6*	150*	530*	1,500*	NA
5		1150S	420	22*	540*	1,900*		
6		1150S*1.8	570	31*	740*	2,600*		
			Duke Rev 2b w RCP seal replaced & flood wall installed					
7	same as above	Duke	14	2*	31*	100*	310*	NA
8		1150S	52	7*	110*	370*		
9		1150S*1.8	70	9*	150*	500*		
McGuire			*Duke PRA Rev 3*					
1	EF=0.26 LF=0.56 NF=0.18 (Duke PRA range)	Duke	13	2*	32*	110*	320*	98
2		1150S	44	8*	110*	380*		340
3		1150S*2.3	72	13*	180*	600*		540

* includes SBO frequency due to tornado

The following assumptions apply to all the cases shown in Table 12:

1. 40 year plant life remaining,
2. 7% discount rate (3% discount rate would increase all results by a factor of 1.74), and
3. late failure is not averted by the enhancement (thus, with the assumptions made for these analyses, onsite health costs are not relevant).

Cases:

<u>Sequoyah 1</u>
For all the Sequoyah cases the SBO frequencies from the NUREG-1150 studies are used, and the consequences are estimated based on the NUREG-1150 source terms, as updated in NUREG/CR-6349 [10], and updated for inflation and population increase. The first case is calculated using the mean early containment failure probability from NUREG-1150.

<u>Sequoyah 2</u>
Same as Sequoyah 1, but using the 95th percentile of the mean early containment failure probability from NUREG-1150.

<u>Sequoyah 3</u>
Same as Sequoyah 1, but using the early containment failure probability from NUREG/CR-6427.

<u>Catawba 1</u>
SBO frequencies are from Rev. 2b of Duke's PRA for Catawba. Note for internal events, the point estimate is strictly internal events, but that the 5th, mean, and 95th values include internal events and tornados. The point estimate for tornados is given separately in the PRA and is only about 10% of the mean (which includes internal events and tornados). Therefore, the inclusion of the tornado events does not have a big effect. Containment failure probability values are within the range for failure probabilities used in the Duke PRA and the same as those in NUREG/CR-6427 for Catawba. The source term person-rem was extrapolated from the health risk information provided in the Duke email, with offsite costs scaled from NUREG-1150 offsite cost estimates based on the comparable person-rem ratios.

<u>Catawba 2</u>
Same as Catawba 1, but using the NUREG-1150 source term/consequence results (i.e., those used in Sequoyah cases above). This was done as a sensitivity case based on the differences shown in Table 9 above.

<u>Catawba 3</u>
Same as Catawba 2, but since the population around Catawba is larger than that around Sequoyah by a factor of about 1.8, the Sequoyah person-rem were increased by that factor.

Catawba 4, 5 & 6

Same as Catawba 1, 2 & 3 respectively, but with the SBO frequencies taking into account reactor coolant pump (RCP) seal replacement. The point estimate for tornados is only about 9% of the mean, so again the inclusion of the tornado events does not have a big effect.

Catawba 7, 8 & 9

Same as Catawba 1, 2 & 3 respectively, but with the SBO frequencies taking into account RCP seal replacement and installation of a flood wall. The point estimate for tornados is about 44% of the mean. Therefore, here the inclusion of the tornado events does have a large effect.

McGuire 1

SBO frequencies are from Rev. 3 of Duke's PRA for McGuire. Again, for internal events, the point estimate is strictly internal events, but the 5th, mean, and 95th values include internal events and tornados. The point estimate for tornados is about 51% of the mean. Therefore, the inclusion of the tornado events does have a large effect. Containment failure probability values are within the range for failure probabilities used in the Duke PRA. The source term person-rem was extrapolated from the health risk information provided in the Duke email, with offsite costs scaled from NUREG-1150 offsite cost estimates based on the comparable person-rem ratios.

McGuire 2

Same as McGuire 1 but using the NUREG-1150 source term/consequence results (i.e., those used in Sequoyah cases above). This was done as a sensitivity case based on the differences shown in Table 2 above.

McGuire 3

Same as McGuire 2, but since the population around McGuire is larger than that around Sequoyah by a factor of about 2.3, the Sequoyah person-rem were increased by that factor.

Note that uncertainties associated with issues such as spontaneous ignition burning off accumulated hydrogen, and less than 100% reliability of the gas control system, would only affect the value of CPEF avoided and, therefore, can be accounted for by varying CPEF. Also note that, aside from the sensitivity calculation with the two different source terms, no uncertainties in the Level 3 part of the calculations involved in the averted cost have been addressed.

It should also be pointed out that the inclusion of averted costs from external events assumes that the combustible gas control system is designed to withstand the external event. For example, the control system would have to be seismically qualified to the appropriate g level to withstand an earthquake of a certain magnitude. Obviously this would increase the cost of the combustible control system above that designed to deal only with internal events.

5. RESULTS FOR A BWR MARK III PLANT

In this Section the benefits accrued from a combustible gas control system which remains functional during SBO sequences are calculated for the Grand Gulf plant, a BWR 6 with a Mark III containment, based on the NUREG-1150 study of Grand Gulf.

The NUREG-1150 study for Grand Gulf was an integrated study (Level 1, Level 2, and Level 3) PRA study and provides, separately, uncertainty ranges for CDF (Level 1) as well as containment failure probability (Level 2). However, only internal events were examined for Grand Gulf in the NUREG-1150 study. Grand Gulf CDF ranges due to SBO events are presented in Table 6.2 of NUREG-1150 Volume 1 [4]. A histogram of early containment failure probability consequential to SBO for Grand Gulf is shown in Figure 2.5-2 of NUREG/CR-4551, Vol.6, Rev 1, Part 1 [6]. Table13 below summarizes the values in the reports.

Table 13 Grand Gulf Uncertainty Ranges for Internal Events

	5th	**mean**	**95th**
SBO CDF frequency from NUREG-1150 (ry)	1.7E-7	3.9E-6	1.1E-5
CPEF due to SBO from NUREG/CR-4551, Vol.6	~1.E-2	~0.5	~1.0

5.1 BWR Mark III Example Benefit Calculations

The benefit calculation for Grand Gulf, using mean values from NUREG-1150, is carried out below following the steps at the beginning of Section 2 of this report.

Step 1 - Frequencies of SBO sequences

As indicated in Table 13, the mean SBO CDF from internal events found in the NUREG-1150 study was 3.9E-6 per reactor-year.

Step 2 - Change in conditional containment failure probability

Considerable information on accident progression and hydrogen deflagration and detonation for Grand Gulf was developed during the NUREG-1150 study and is documented in NUREG-1150 and the supporting documents [4,6]. This information is summarized in Reference 8 and the following discussion is based on Reference 8.

Mark III containments depend on glow plug hydrogen igniters to control pressure loads resulting from hydrogen combustion events. If the igniters are not operating, due to lack of AC power (the dominant sequence being an SBO) or operator failure to manually actuate them, there is a possibility of an energetic hydrogen combustion (deflagration or detonation) event at the time of vessel failure (or at other times if the operators fail to follow procedures and the igniters are actuated when a significant amount of hydrogen has accumulated). These energetic combustion events were stated in NUREG/CR-1150 and the supporting documentation for Grand Gulf (NUREG/CR-4551, Volume 6 [6]) to result in early containment failure with a relatively high conditional probability (~0.5). However, in a Mark III containment an unscrubbed release (one which does not pass through the suppression pool) requires failure of the drywell in addition to containment failure. Drywell failure can occur: (1) directly as a result of loads associated with vessel breach or from hydrogen combustion, or (2) indirectly as a result of structural failure of the pedestal.

Before vessel breach, the only significant event that was found in NUREG/CR-4551, Volume 6, to cause drywell failure was hydrogen combustion in the wetwell. However, at the time of vessel breach loads from direct containment heating, ex-vessel steam explosions, hydrogen combustion, and RPV blow down contribute to the probability of drywell failure. Accordingly, loads from high pressure vessel breach and hydrogen combustion were determined to be the leading causes of containment and drywell failure.

The Grand Gulf (NUREG/CR-4551, Volume 6) results are summarized in Table 14 below. This table indicates that accident sequences that contribute to large releases (which require failure of the drywell in addition to containment failure) are sensitive to the type of accident (i.e., SBO vs non-SBO) and the pressure (i.e., transient vs large break LOCA) in the reactor pressure vessel at the time of vessel breach.

Table 14: Conditional Containment and Drywell Failure Probabilities for Mark III Containments

Reactor Coolant System (RCS) Pressure at Vessel Breach	Station Blackout, SBO (Igniters and Sprays unavailable)		Non-SBO (Igniters and Sprays available)	
	Containment Fail	Containment and Drywell Fail	Containment Fail	Containment and Drywell Fail
High	~ 0.5	~ 0.2	~ 0.5	~ 0.2
Low	~ 0.5	~ 0.2	~ 0.01 - 0.02	~ 0.01

As shown in the table, if the RCS is at high pressure the likelihood of containment failure is relatively independent of whether or not the igniters are operating. In addition, the likelihood of simultaneous failure of the drywell is also independent of igniter operation if the RCS is at high pressure.

As the above table indicates, if the RCS is depressurized at vessel breach, the likelihood of containment failure is dependent on whether or not the igniters are operating. If the igniters are not available, the conditional probability of containment failure is approximately 0.5 even with the RCS at low pressure. The likelihood of simultaneous failure of the drywell is also about 0.2 at the time of vessel breach. Thus, all SBO sequences (without combustible gas control) have a conditional probability of 0.2 of a large release, regardless of the pressure in the RCS.

The potential for containment failure at the time of vessel breach when the RCS is at low pressure and the igniters are operating is not directly assessed in NUREG/CR-4551, Volume 6. However, the conditions prior to vessel breach should be applicable to this situation because the RCS is depressurized and none of the issues associated with high pressure melt ejection would occur. The results prior to vessel breach indicate a conditional probability of containment failure in the range of 0.01 to 0.02 if the igniters are operating.

In summary, for transient sequences with the RCS at high pressure and for all SBO sequences the conditional probability is close to 0.2 that the Mark III containment fails at the same time that the suppression pool is bypassed. However, if the RCS is depressurized and the igniters are operating, then the conditional probability is less than 0.1 that the Mark III containment will fail. The IPE database (www.nrc.gov/NRC/NUREGs/SR1603/index.html) information on the plant damage states (PDSs) for the four domestic Mark III plants was searched to determine the fraction of PDSs that have low RCS pressure. The average across the four plants for PDSs with this attribute is approximately 40 percent, with high RCS pressure making up the remaining 60 percent.

Based on Table 14 and the above discussion, the following event tree can be constructed and quantified, conditional on an SBO event without a hydrogen control system operating. The late failure split fractions are based on NUREG-4551 Vol. 6 results.

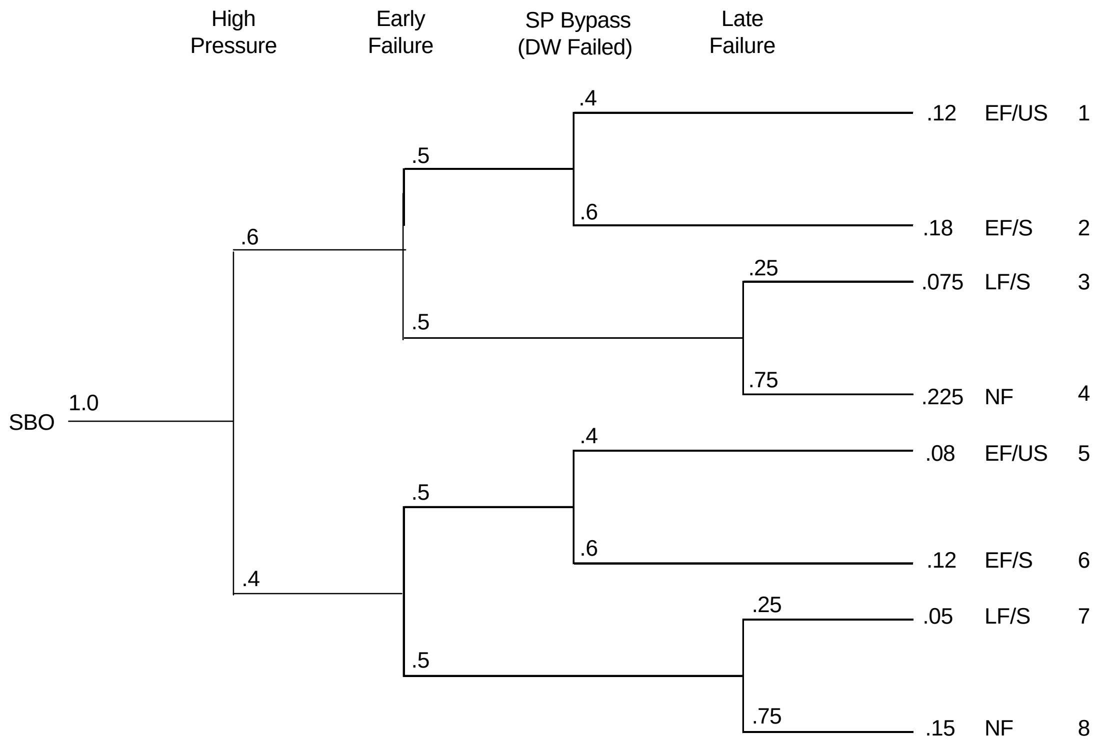

Figure 1: Containment event tree conditional on SBO without combustible gas control

The top events are high RCS pressure, early containment failure, drywell failure, and late containment failure. A late containment failure will always be scrubbed. The conditional probability for each of the 8 end states is shown in the figure. EF, LF, and NF indicate early containment failure, late containment failure, and no containment failure, respectively. US indicates an unscrubbed release, S indicates a scrubbed release.

A similar event tree, based on Table 14 and the accompanying discussion, can be constructed for SBO events assuming combustible gas control is still functional. This event tree is shown in Figure 2. (Note that the 1.0/0.0 split fraction on the low pressure branch SP Bypass event is chosen for conservatism, and has very little effect on the results.)

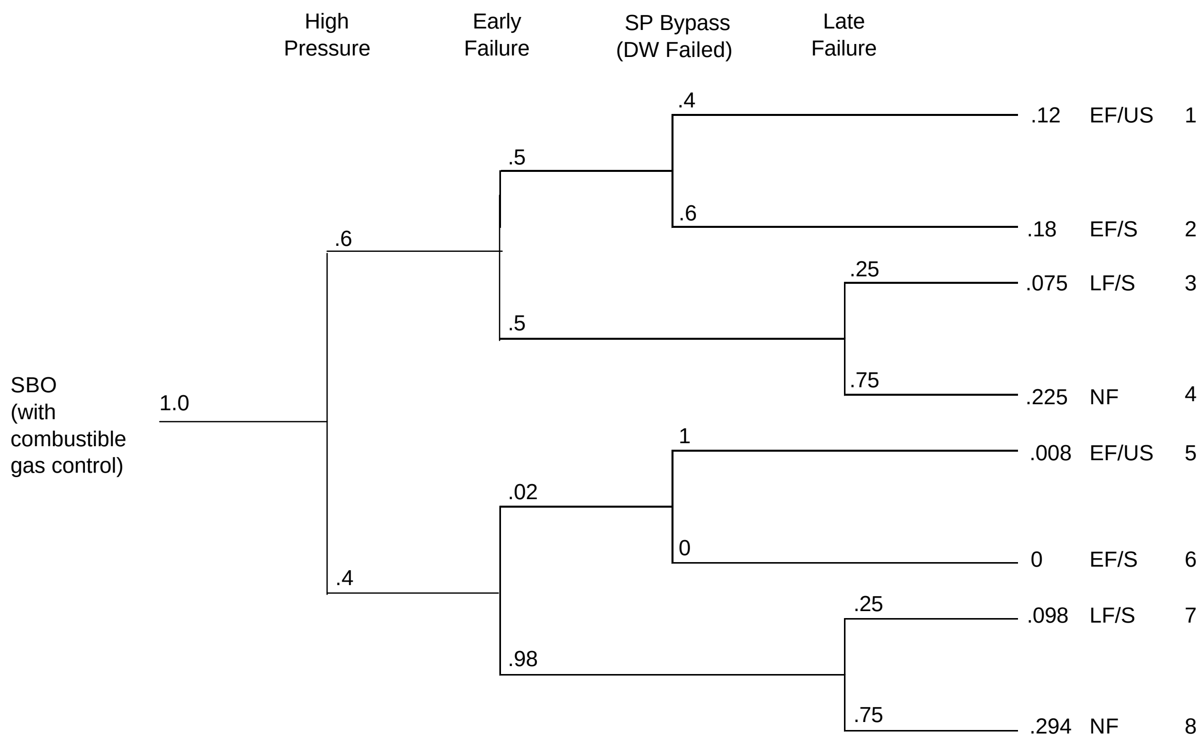

Figure 2: Containment event tree conditional on SBO with combustible gas control functional

A comparison of the trees shows that the high pressure, i.e., upper half of both trees, is identical. This means that any benefit gained from a combustible gas control system which functions during SBO will depend only on the different conditional probabilities associated with low pressure scenarios (end states 5 through 8).

Step 3 - Consequences associated with each containment failure mode

Offsite consequences for releases at Grand Gulf representative of each of the end states indicated in Figures 1 and 2 are shown in Table 15. No consequences are assumed for no containment failure. Offsite person-rem and offsite property cost estimates are based on the data provided in References 10. These results are conditional consequences (i.e., conditional on occurrence of the release) out to 50 miles from the plant and include offsite population dose (person-rem) and offsite damage costs.

Two values for offsite person-rem are shown here as well. The 1990 values are based on Reference 10. The 2000 values have been updated based on the change in population density from 1990 to 2000 as estimated in the Grand Gulf FSAR. The change is an increase of about 7%.

Two values are also shown for the offsite property damage costs. The first is taken from Reference 10 and is in 1990 dollars. The second updates the 1990 dollar values to current year dollars based on the price inflation calculator (approximately 36% over the 1990-2002 period) of the U.S. Bureau of Labor Statistics (www.bls.gov).

Table 15: Offsite Consequences (50-mile radius) of Containment Failure Releases at Grand Gulf						
Sequence	Failure Mode	Offsite Person-rem 1990	Offsite Person-rem 2000	Offsite Health Effects $k	Offsite Property 1990$k	Offsite Property 2002$k
GG-11-1	Early unscrubbed	5.7E+05	6.1E+05	1,200,000	810,000	1,100,000
GG-04-1	Early scrubbed	1.0E+05	1.1E+05	220,000	43,000	59,000
GG-18-1	Late scrubbed	7.0E+04	7.5E+04	150,000	11,000	14,000

GG-11-1 from Reference 6 is a typical early failure unscrubbed sequence with about 99% of noble gases, 38% of iodine, 14% of cesium, and 9% of tellurium released. GG-04-1 is a typical early failure scrubbed sequence with about 76% of noble gases, 5% of iodine, >1% of cesium, and negligible amounts of tellurium released. GG-18-1 is a typical late failure scrubbed sequence with about 83% of noble gases, 1% of iodine, and negligible amounts of cesium and tellurium released.

Again, it should be noted that the (1990) consequences reported in Reference 10 differ somewhat from those reported in the NUREG-1150 reports, even though Reference 10 is based on the NUREG-1150 analyses. This is primarily because in the NUREG-1150 study the consequence analysis was carried out using Version 1.5.11 of the MACCS code, while the consequences in Reference 10 were recalculated with Version 1.5.11.1 of MACCS. This later version explicitly incorporates the higher BEIR V risk coefficient for the latent cancer-dose relationship while the earlier version of MACCS used the BEIR III risk coefficient. In addition, a few input errors in the NUREG-1150 MACCS calculations were corrected for the recalculations of Reference 10.

Onsite health consequences again are calculated assuming 20,000 person-rem occupational exposure, or $40,000k after using the $2000/person-rem factor, for all early and late containment failures, and 8,000 person-rem, or $16,000k, for no containment failure. Onsite property damage is not included as per the discussion in Section 3.

Step 4 - Summation of conditional containment failure modes and their consequences

The results of the summation of conditional containment failure modes and their consequences are shown in Table 16.

Table 16: Summation of Offsite Costs and Onsite Health Effect Costs		
Gas Control	Total Offsite Cost Conditional on SBO ($k)	Onsite Health Effects Cost Conditional on SBO ($k)
no	570,000	31,000
yes	380,000	28,000

Step 5 - Subtraction of costs and multiplication by frequency

The calculation in Step 4 was made with and without the gas control system present. The difference between the cases where gas control is 'yes' and the cases where gas control is 'no,' when multiplied by the SBO frequency, represents the averted offsite cost on a per reactor-year basis. The results are summarized for Grand Gulf in Table 17 below. Costs are divided into offsite and onsite costs, as well as total costs. Offsite costs are the dominant contributor in all cases. Costs are in 2002 dollars.

Table 17: Cost Summary per reactor-year for Grand Gulf (Internal Events)			
SBO frequency	Total Averted Offsite Costs $k per reactor-year	Averted Onsite Health Effects Costs $k per reactor-year	Total Costs $k per reactor-year
3.9E-6	0.76	.014	0.77

Step 6 - Calculation of lifetime benefit

Multiplication by the present worth factor, based on the discount rate selected and plant lifetime remaining, yields the total averted offsite cost, or benefit, over the plant's lifetime. Results for a lifetime of 40 years for a discount rate of 7% and 3% are shown in Tables 18 and 19, respectively. This step completes the analysis.

Table 18: Lifetime Benefit Base Case (7% Discount Rate) for Grand Gulf			
Internal Events	Lifetime Averted Offsite Costs 2002$k	Lifetime Averted Onsite Health Effects Costs 2002$k	Lifetime Total Costs Averted 2002$k
	10k	0.18	10k

Table 19: Lifetime benefit sensitivity case (3% discount rate) for Grand Gulf			
Internal Events	Lifetime Averted Offsite Costs 2002$k	Lifetime Averted Onsite Health Effects Costs 2002$k	Lifetime Total Costs Averted 2002$k
	18	0.3	18

The results are again dominated by the offsite costs but are much smaller than for the ice condensers. For Grand Gulf the total averted offsite costs due to internal events amount to $10k for a 7% discount rate and $18k for a 3% discount rate.

Inclusion of averted onsite costs produces a negligible change in all cases. Since the Mark III containments considered here are single nuclear units, the discussion of Section 3 regarding onsite costs related to the effect of containment failure would imply that onsite property damage costs averted by adding a combustible gas control system which functions under SBO conditions would also be small.

5.2 BWR Mark III Uncertainty Considerations

To estimate the uncertainty in benefits achieved by enhancing gas control in BWR Mark III containments to operate under SBO conditions, BNL:

1. made additional benefit estimates based on the uncertainty results from the NUREG 1150 study for Grand Gulf,
2. ran the latest available SPAR model for Grand Gulf and River Bend and calculated benefits based on the uncertainty in the SBO frequencies provided in these models, and
3. reviewed the IPEs and IPEEEs for variation in SBO CDF and variation in CPEF for Mark III plants.

No recent industry PRAs, similar to those made available for the ice condenser plants, were available for the Mark III benefit estimates.

NUREG-1150 Grand Gulf

Table 13 above summarized the 5th percentile, mean, and 95th percentile values for both SBO CDF and the CPEF found for Grand Gulf in the NUREG-1150 study.

A series of benefit calculations were made using the NUREG-1150 SBO frequencies and the accident progression scenarios from Figures 1 and 2, above. The results of the calculations are summarized in Table 22 below. Benefits were estimated with the split fractions in Figures 1 and 2 (which assume the NUREG-1150 mean value for CPEF) for the 5th percentile, mean, and 95th percentile NUREG-1150 SBO frequencies (Grand Gulf 1 in Table 22).

To further examine the uncertainty in benefits, a sensitivity calculation was made using the 95th percentile for CPEF, which is essentially 1.0, i.e., the containment always fails (Grand Gulf 2 in Table 22). This assumption will increase the benefit from gas control during SBO.

Another sensitivity calculation was made to further increase the benefits by assuming half (rather than 40%) of all sequences are at low pressure, and assuming drywell failure occurs whenever containment fails (Grand Gulf 3 in Table 22). This is quite a conservative case and should provide some reasonable upper bound on the benefit.

Since benefits are already low in the base case, no lower range sensitivity calculation was carried out.

SPAR Model Runs

To further estimate benefits as well as the uncertainty associated with the Level 1 PRA calculations, BNL ran the latest available 3i SPAR model for Grand Gulf, an internal events, Level 1 model, which incorporates uncertainty parameters and can calculate a point estimate, the mean, median and various percentiles associated with the SBO CDF. The model incorporates up-to-date information on loss of offsite power frequency and emergency diesel generator availability. Similar to the ice condenser models, these Mark III SPAR models have not undergone a quality assurance process as yet, and the model software warns the user that the 3i versions are developmental versions that have not been peer reviewed, may contain errors, and may change. However, since no up to date Mark III PRAs were made available for the benefit estimates, the results with the SPAR model frequencies are included here. The NUREG-1150 accident progression was again assumed, and the same sensitivity cases were run. The results are illustrated in Table 22 (Grand Gulf 4, 5, 6).

In addition, the 3i SPAR model for River Bend was also exercised and benefit results were obtained, again using the NUREG-1150 Grand Gulf accident progression scenario for the Level 2 analysis. For the consequence calculations, the NUREG-1150 Grand Gulf person-rem values for all sequences were increased by a factor of 3.1 to account for the increased population density around River Bend. Benefits were again calculated for the base case of the accident progression split fractions of Figures 1 and 2 and the two sensitivity cases (River Bend 1, 2, 3, respectively in Table 22). SPAR model SBO frequencies are shown in Table 20.

Table 20 SPAR 3i SBO CDF ranges for internal events (ry)

	5th	**mean**	**95th**
Grand Gulf	1.4E-7	2.4E-6	8.2E-6
River Bend	2.7E-8	1.0E-5	2.8E-5

The uncertainty associated with the Level 2 calculations for Grand Gulf cannot be estimated with the SPAR models, since no Level 2 SPAR models incorporating uncertainty are available.

IPE and IPEEE Comparisons

The PRAs conducted for the IPE Program and the IPEEE Program did not include uncertainty estimates. However, a survey of the SBO frequencies and containment failure probabilities used in the IPE and IPEEEs was carried out for this report and the results are shown in Table 21, including some of the reasons for the variation in frequency.

Table 21 SBO Frequencies from the IPEs (ry)

Plant	Internal Events	External Events	Additional information from IPEs
Clinton	9.8E-6	not available	Separate standby service water (SSW) system for emergency loads. For LOSP uses high initiating event and non-recovery frequency.
Grand Gulf	7.5E-6	not available	Separate SSW system for emergency loads. SSW pump room ventilation failure an important contributor.
Perry	2.2E-6	not available	Only Mark III to credit fire water for injection early in SBO sequences.
River Bend	1.4E-6	not available	SSW failures lead to short term SBO Credits prevention of switchover to high temp suppression pool to keep reactor cooling isolation system (RCIC) working.

As Table 21 indicates, the internal events SBO CDFs for Mark III plants in the IPEs are well within the range (5^{th} to 95^{th} percentiles) of the Grand Gulf NUREG-1150 SBO frequency and the SPAR model frequencies. Note that the River Bend IPE frequency is an order of magnitude lower than the 3i SPAR model frequency. No external event frequencies are available for Mark III plants from the IPEEEs.

Variation in population density around the plant sites was also surveyed. Based on FSAR projections, Perry has the highest projected year 2000 (50 mile radius) population density, about 7.5 times that of Grand Gulf, which has the lowest. Both Clinton and River Bend have population densities that are about 3.1 times that of Grand Gulf.

Although Perry has the highest population ratio, it also has the lowest SBO frequency. Therefore, since the estimates for River Bend were done with the (high) SPAR 3i model SBO frequencies and by accounting for the increased population density around River Bend (vs. Grand Gulf), the River Bend calculations (River Bend 1, 2, 3, in Table 22) should provide a bound for all four Mark III sites.

5.3 Summary of BWR Mark III Results

Table 22 summarizes the results of the calculations carried out for estimating the benefit of an enhanced combustible gas control system for the BWR Mark III plants. Note that no uncertainties in the Level 3 part of the calculations involved in the averted cost have been addressed.

Table 22 Averted Costs ($k)

Plant & Case description		Source of SBO frequency			
		Internal Events			External Events
		5^{th}	mean	95^{th}	
Grand Gulf		*NUREG-1150*			
1	Mean NUREG-1150 CPEF Split fractions from Figs 1&2	<1	10	29	NA
2	95^{th} NUREG-1150 CPEF Split fractions from Figs 1&2	<1	22	61	
3	95^{th} NUREG-1150 CPEF 50% of sequences at low pressure, drywell always fails if containment fails	2	60	170	
		SPAR 3i			
4	Mean NUREG-1150 CPEF Split fractions from Figs 1&2	<1	6	22	NA
5	95^{th} NUREG-1150 CPEF Split fractions from Figs 1&2	<1	13	45	
6	95^{th} NUREG-1150 CPEF 50% of sequences at low pressure, drywell always fails if containment fails	2	36	120	
River Bend		*SPAR 3i*			
1	Mean NUREG-1150 CPEF Split fractions from Figs 1&2	<1	57	160	NA
2	95^{th} NUREG-1150 CPEF Split fractions from Figs 1&2	<1	120	330	
3	95^{th} NUREG-1150 CPEF 50% of sequences at low pressure, drywell always fails if containment fails	<1	320	880	

The following assumptions apply to all the cases shown in Table 22:

1. 40 year plant life remaining
2. 7% discount rate (3% discount rate would increase all results by a factor of 1.74)

6. DISCUSSION OF RESULTS

Comparison of the results in Section 4 for the PWR ice condenser plants with the results in Section 5 for the BWR Mark III plants shows that the estimated benefit of providing combustible gas control during SBO sequences differs significantly for these two plant types. Using lifetime averted offsite costs for internal events for the example case, i.e. the mean NUREG-1150 case (7% discount rate), the Sequoyah (ice condenser) cost estimate (with late failure) is $320k, while the Grand Gulf (Mark III) lifetime averted costs for the mean NUREG-1150 case is estimated at $10k. In other words, the Sequoyah results are higher than the Grand Gulf results by a factor of roughly 30.

The reasons for this large difference can be attributed to a number of factors involved in the analyses of these plants:

1. The SBO frequency is lower for Grand Gulf
2. The CPEF averted by the combustible gas control system is lower for Grand Gulf (and Mark III's in general) because
    (a) the early failure of both the containment and the drywell are necessary to obtain significant consequences, and
    (b) the igniters are assumed effective only for low pressure sequences.
3. The conditional offsite person-rem are lower for Grand Gulf.

Comparison of these parameters is illustrated in Table 24 below.

Table 24: Parameter Comparison

Parameter	**Sequoyah Value**	**Grand Gulf Value**	**Sequoyah/Grand Gulf**
SBO frequency	1.5E-5	3.9E-6	3.8
Approximate averted CPEF*	0.15	0.09	1.7
Offsite person-rem 2000 estimate	3.1E+6	6.1E+5	5.1
TOTAL FACTOR			~30

*CPEF: for Grand Gulf the value shown is a weighted (by consequences) average of the CPEF averted in end states 5 and 6 of Figure 2.

7. REFERENCES

1. US NRC, "Risk-Informing 10 CFR 50.44," SECY-00-0198, December 2000.

2. US NRC, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," NUREG/BR-0058, Rev. 3, Final Report, July 2000.

3. US NRC, "Regulatory Analysis Technical Evaluation Handbook," NUREG/BR-0184, Final Report, January 1997.

4. US NRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, Final Summary Report, December 1990.

5. J.J. Gregory, et al., "Evaluation of Severe Accident Risks: Sequoyah, Unit 1," NUREG/CR-4551, Vol. 5, Rev. 1, Part 1, Main Report, December 1990.

6. T.D. Brown, et al., "Evaluation of Severe Accident Risks: Grand Gulf, Unit 1," NUREG/CR-4551, Vol. 6, Rev. 1, Part 1, Main Report, December 1990.

7. M.M. Pilch, et al., "Assessment of the DCH Issue for Plants with Ice Condenser Containments," Sandia National Laboratories, NUREG/CR-6427, February 2000.

8. W.T. Pratt and V. Mubayi, "Basis Document for Large Early Release Frequency (LERF) Significance Determination Process (SDP)," NUREG/CR-xxxx, (to be published)

9. Communication from H. Duncan Brewer, Duke Power, via email, September 20, 2002.

10. V. Mubayi, et al., "Cost-Benefit Considerations in Regulatory Analysis," Brookhaven National Laboratory, NUREG/CR-6349, October 1995.

11. Richard P. Burke, David C. Aldrich, and Norman C. Rasmussen, "Economic Risks of Nuclear Power Reactor Accidents," Sandia National Laboratories, NUREG/CR-3673, April 1984.

12. D.R. Strip, "Estimates of the Financial Consequences of Nuclear Power Reactor Accidents," Sandia National Laboratories, NUREG/CR-2723, September 1982.

13. Bureau of Labor Statistics website www.bls.gov.