Part B:
What impact will long-term exposure to low dose, low-dose-rate radiation have on human health?

Throughout the world our estimates of the risks of radiation are based on close examination of the survivors of the A-bomb attacks of 1945[2]. Risk estimates so derived are used to project the long-term effects of *any* exposure to man-made radiation. They are used in the setting of dose limits for occupational exposures or exposure of the general public, for setting Protective Action Guidelines following accidental or intentional (weapons) radiation release, and for setting 'return-home' guidelines, as encountered here.

B.1 The A-bomb Survivors Dataset:

Radiation exposure of the inhabitants of Hiroshima and Nagasaki in August 1945 led to doses ranging from very small to very large, depending on distance from ground zero, but in all cases the dose delivery was very rapid. Because of the height of the blast (1800-1900 ft above the earth), little material from the ground was taken up into the fireball, leading to low levels of radioactive fall-out [23]. Almost all of the radiation dose to the population can be considered as arising from prompt bomb radiation, that is, within the first minute. Long term, low dose rate radiation exposure due to radionuclides in the environment was not experienced.

The effects of large radiation doses were observed within days to months as some of the survivors of the bomb's blast and thermal effects died of symptoms of the acute radiation syndrome. Later, based on results of questionnaires associated with the 1950 Japanese national census [23], 93,700 individuals exposed to A-bomb radiation and 26,600 unexposed persons (residents located more than 18 km from ground zero) were identified and entered into what is now called the Life Span Study (LSS). This study has been ongoing for six decades and is the second longest running epidemiological study of health effects (after the Framingham Heart study) ever conducted.

Individual dose determination for each LSS participant began with a personal interview to determine precise location and body orientation at the time of the blast [24]. Radiation transport calculations then track neutron and gamma fluence from the fireball through various shielding structures between the individual and ground zero. These calculations have improved considerably over the past few decades as a result of greater sophistication in the modeling, newly available interaction cross-section data for important isotopes, finer energy group structure for particle transport, and increased confidence in the dosimetry models resulting from detailed comparisons between model predictions and measured data. Photon fluence estimates at varying distances from ground zero have been compared with thermoluminescence measurements [24] in tiles and bricks that were exposed to gamma-rays from the bomb; neutron fluence calculations have been compared with neutron-induced radioactivity [25,26] in tile, granite, concrete, and soil samples[3]. The ability to compare model-driven fluence estimates with actual measurements taken in various locations and distances from ground zero has resulted in iterative refinement of neutron and photon dose estimates. The doses that individual survivors received are thus now known to a good level of accuracy [24].

Individuals in the LSS are followed until death, and the cause(s) of death recorded. Cancer fatality rates in the exposed and control groups are compared[4] and the relative excess is plotted versus the dose received. Figure 2 is taken from Preston et al (2004), a study that applied the results of the latest dosimetry analysis (DS02) to the cancer death rates observed in the Life Span Study participants; plotted is excess risk of radiation-induced solid cancers (Figs. 2a and 2b) or leukemia (Figs. 2c and 2d) versus dose [27].

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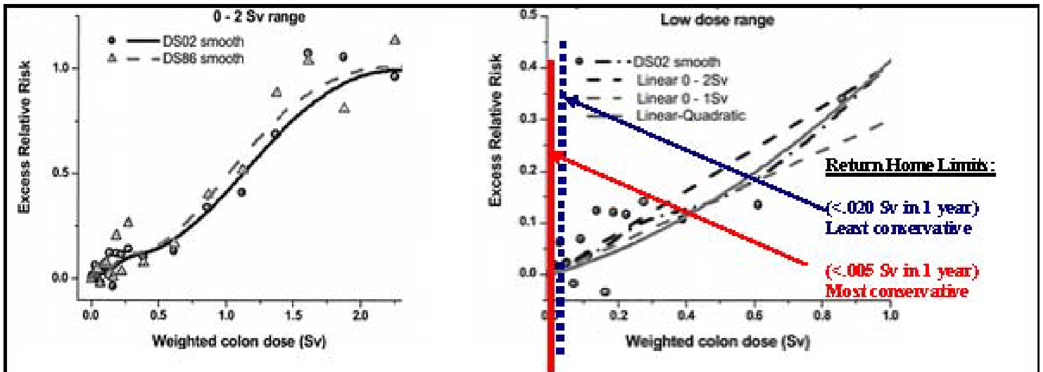
[2] This is the largest and most general population exposed to radiation over a wide range of doses. It is also the largest single cohort of generally healthy individuals exposed to radiation as children.

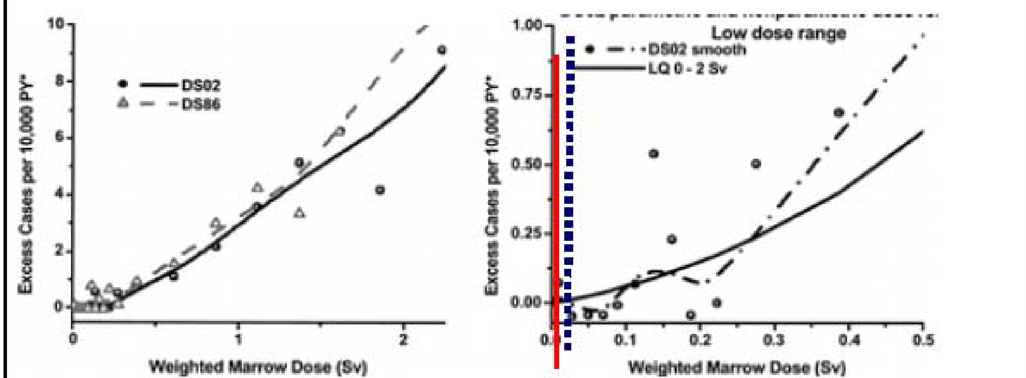
[3] ^{59}Co (^{60}Co, $T_{1/2}$ = 5.3 yr), ^{151}Eu in tile and granite samples (^{152}Eu, $T_{1/2}$ = 13.5 yr), and ^{35}Cl in concrete (^{36}Cl, $T_{1/2} = 3 \times 10^5$ yr); $^{63}Cu(n,p)^{63}Ni$ ($T_{1/2}$ 101 yr).

[4] Health effects other than cancer have been examined at high doses; however at low doses non-cancer risks are especially uncertain, according to BEIR VII [7], and are not typically incorporated into risk estimates. Radiation induced mutations in sperm or ova resulting in heritable disease are so low in risk as to be undetectable in humans, even in A-bomb survivors [7].

The dose axes span a very large range (for instance, 2.0 Sv is the equivalent of ~700 times the average natural background dose in the U.S. but delivered within one minute). At large doses (>0.5 Sv) it is clear that survivors are at an increased risk of dying of cancer in later life. At the lowest doses (below ~ 0.1 Sv), the large natural cancer rate in both the exposed and the unexposed populations makes it impossible to declare, with any certainty, what effect small radiation doses have on the cancer fatality rate in exposed persons. But it is precisely this low dose range that we are most interested in. And, since this is the best population available for examining radiation-related health effects, we must use these data in some way to provide guidance for the safe conduct of human activities involving the use or production of ionizing radiation.



(a) Excess relative risk of solid cancer mortality. (b) Low dose range of plot (a).



(c) Excess absolute risk of leukemia mortality. (d) Low dose range of plot (c).

Figure 2. Excess cancer risk versus dose. Plots on the left show data points using the prior dosimetry system DS86 (triangle) and the new (DS02) dosimetry calculations (circles). An enlargement of the lower dose data is shown on the right side. Shown are the DS02 data points along with linear fits based on the dose ranges 0 – 1 Sv and 0-2 Sv and a linear quadratic fit based on the 0-2 Sv dose range. [From D.L. Preston et al, "*Effect of Recent Changes in Atomic Bomb Survivor Dosimetry on Cancer Mortality Risk Estimates,*" **Radiation Research** (162) 377-89, 2004 [27].]

In 1(b) and (d) the the RHDLs are superimposed on the low-dose cancer risk plots. The vertical lines representing the Pennsylvania (5 mSv) and the EPA RHDL (20 mSv) are the doses accumulated over the first year of return-home habitation. [This is an important distinction from the A-bomb doses referred to on the plot's abscissa which were delivered in less than a minute. See more on this point in Part B.3.]

Making Use of the A-bomb Survivor Dataset: To make use of the A-bomb survivor data we must assume some shape of the risk versus dose relationship, then fit an equation to this shape, and use this equation to estimate the risks associated with doses far lower than doses at which statistically significant estimates of risk exist based on the data alone. The equation usually used is a straight line that passes through the origin (0,0). When constrained to pass through zero at the low dose end, the slope of the curve is driven by the risk associated with high doses. With this "linear, no threshold" (LNT) approach, a risk-per-unit-dose estimate is generated (the slope of the line) that can be easily applied throughout the entire dose range. Most agencies and committees involved in evaluating available data and generating risk estimates for ionizing radiation use this approach.

From Risk Estimate to Occupational Dose Limits: This estimate of risk-per-unit-dose is widely used. It is used, for instance, to derive dose limits for workers in occupations dealing with exposure to radiation (after modification for dose-rate effects; see below). Occupational dose limits are set using the "safe industries" argument [28]. The risk to a radiation worker of dying from a job-related cause must be no higher than the risk encountered by workers in safe industries. Using the risk-per-unit dose estimate generated by the LNT approach, the dose that would generate a risk equivalent to that experienced by a worker in a safe industry (on-the-job accidents in retail, government, manufacturing, etc.) is determined and the dose limit is set accordingly.[5] This limit is set at 50 mSv per year.

From Risk Estimate to Public Dose Limits: Dose limits recommended for the worker are reduced by a factor of 10 for application to protection of the public [28]. There are several justifications for setting the limits lower than those for radiation workers. First, the public does not directly benefit (in a wage-related manner) from exposure to the anthropogenic radiation. Second, a wider range of sensitivities is expected to be found in the general population than the adult worker population, and third, the period of exposure can potentially be longer. Dose limits to the public are typically comparable to local variations in natural background radiation [13]. The limit of dose to the public from all anthropogenic radiation sources (excluding medical) is set at 5 mSv per year.

These dose limits (or very similar) to radiation workers and to the public are applied as part of standard radiation protection around the world. They in fact represent minimum standards since every radiation facility is also required to meet the standards of ALARA. Thus radiation workers in nuclear power plants in the US rarely come close to meeting the 50 mSv limit, and the average annual dose to a nuclear power plant worker, steadily declining over the years, is 1.2 mSv (120 mrem) [29]. Similarly, members of the public rarely receive doses at or even near the 5 mSv dose per year limit (with the exception, of course, of their medical doses).

From Public Dose Limit to RHDLs: The EPA recognizes that the recommended upper bound for dose to the public from manmade sources in a single year (5 mSv) was not developed for nuclear incidents and is not appropriate for chronic exposure [1]. They recommend a RHDL of 20 mSv in the first year and 5 mSv thereafter. These RHDLs are shown superimposed on Figures 1b and 1d.

B.2 The accident scenario requires a different approach than for routine radiation protection.

The setting of RHDLs (and other protective action guidelines) at values similar to dose-limits used for routine radiation protection (within factors of several) is problematic for a number of reasons.

First, for decades, all of our risk estimates have been generated for the express purpose of protecting workers and the public from man-made sources of radiation. Because there are few data directly relevant to the impact of low dose-rate radiation, several extrapolations from actual data must be applied. Conservative assumptions, clearly stated, are applied throughout the process. This puts additional onus on the manufacturer/user of the radiation to implement a rigorous system of protection that ensures keeping the dose to workers and the public always very

[5] Note that cancer deaths (including radiologically-induced cancers) occur late in life unlike industry deaths from other causes which lead to an average age of death at 40 years. Thus, radiation leads to fewer years of life lost than with deaths in other industries [28].

low. If the manufacturers cannot maintain these standards of safety they can no longer operate in which case the public continues to be protected. Since doses are, for the most part, very low, the lack of actual data concerning the harm of somewhat higher doses does not become an issue.

However, once elevated radionuclide levels exist in the environment it is not conservative estimates that are needed but actual best estimates of the hazard the public will face if certain dose avoidance efforts are not undertaken. In the post-accident situation, weighty decisions will need to be made regarding: continuing to live in one's home vs. staying away for a long period of time, perhaps forever; allowing vs. prohibiting access to public roads and buildings; selling vs. destroying local produce and livestock [1], etc. The most accurate estimate of the harm generated by the actual radiation exposure conditions is what the public will need at this time.

Second, if we consider the origins of the return-home dose-limits, we have the following steps:

- Using a linear extrapolation of the A-bomb data down to low doses we generate an estimate of the risk of cancer fatality per Sievert.
- We determine the radiation dose needed to result in a risk of death that matches the risk of on-the-job fatality for workers in safe industries such as trade, retail, and government; this dose becomes the maximum permissible dose to workers (50 mSv).
- We then divide the occupational dose limit by 10 to serve as the dose limit to members of the public (5 mSv).
- We then multiply this by 4 (20 mSv first year RHDL, EPA) or not at all (5 mSv per year RHDL, PA) in response to an unexpected or emergency-related release of radionuclides in the environment.

Thus, the public is not permitted to return to their homes until the risk they are exposed to from residual radiation has dropped to 40% or 10% of the risk a government worker faces from an on-the-job fatality. It is likely that most people would consider this level of risk to be minor. On the other hand, permanent or even long-term removal from one's home would be considered a major event. It is unlikely that members of the public would consider this an acceptable trade-off.

Third, the limited data we do have regarding health effects of elevated dose-rate do not support handling routine radiation protection scenarios and those involving long-lived radionuclides in the environment in a similar way. In fact, the data support a different approach for each scenario. The difference comes in assigning the dose-rate effectiveness factor (DREF).

B.3 The Dose Rate Effect:

At average background dose levels in the US, each cell in our body is traversed ("hit") by a secondary electron from natural radiation approximately once each year [30,31]. Around the world, however, there is considerable variation in soil and rock composition leading to dose-rates ranging to more than 100 times the US average [15]. So, the cells of many people around the world experience many more hits per year, as many as 2-3 hits per week.

The doses to the A-bomb survivors were received in less than 1 minute. For those receiving, say, a 2 Sv dose, each cell in their bodies, on average, would have been hit ~700 times, all within 60 seconds. These variations in dose-rate are illustrated on Figure 2 which shows a logarithmic scale of dose-rate in units of both cSv/min and average number of hits/cell per unit time. Superimposed on Figure 2(b) are the return-home dose-rates for Pennsylvania (most conservative) and Virginia/EPA (least conservative). Note the very large difference (at least a factor of 100,000) between the dose-rates encountered by the A-bomb survivors and those we will encounter when returning home, post-accident.

It matters to our bodies how quickly the dose is received. Whether it comes all at once or spread out over a year makes a big difference. This difference is referred to as the "dose-rate effect". This effect has been studied extensively but in general, it has only been studied at those dose-rates that produce an observable biological effect in laboratory studies. The problem is, dose-rates producing a measurable, or even noticeable effect tend to be orders of magnitude greater than those represented by the RHDLs.

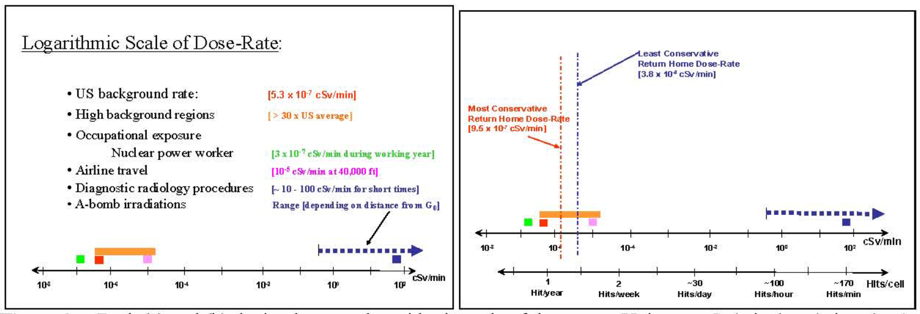


Figure 2. Both (a) and (b) depict the same logarithmic scale of dose-rate. Units are cSv/min (rem/minute). An additional scale is provided on (b) and shows number of radiation tracks (hits) per cell in the body per unit time (assuming 1 hit/cell from an average whole body dose of 3 mSv [30,31]). Shown in (a) are dose-rates from a variety of natural and anthropogenic radiation sources. In (b) the dose-rates corresponding to the return-home dose-rates, averaged over the first year, are superimposed on the data provided in (a).

NCRP 64 "Influence of Dose and its Distribution in Time on Dose-Response Relationships for Low-LET Radiations" (1980) investigated a dose-rate effectiveness factor (DREF) to be used in situations when radiation dose is delivered over time as opposed to delivered acutely [32]. The DREF allows us to take the risk estimates generated using the A-bomb survivor data and apply them to occupational exposures.

In examining the available data, NCRP 64 distinguished between two irradiation scenarios: "protraction" and a "true dose-rate effect". Protraction refers to:

> "exposure times constituting a significant or sizeable fraction of the life span.... Long enough to permit age-dependent changes in the radiosensitivity of the target (e.g. changes in susceptibility to tumor induction or expression with age)" [32].

On the other hand, the true dose rate effect describes shorter-term exposures not influenced by factors important in protraction effects (i.e. includes effects of DNA repair but not of age) [32].

Published data from laboratory studies examining dose-rate effects were examined. Longer term exposures were evaluated separately from shorter-term exposures allowing for separate determination of "protraction factors" (PF) and "dose-rate effectiveness factors" (DREF). Each factor was estimated by fitting high dose-rate and low dose-rate data to linear relationships, both of which were constrained to pass through (0,0), and then taking the ratio of the slopes of these lines.

Clear differences in magnitude were noted between PFs and DREFs. Protraction Factors for cancer induction generated by long-term irradiation ranged from 6.6 to 12.8 with a mean of 10. That is, long-term, low dose-rate radiation is approximately 10 times less effective in cancer induction than the same dose delivered acutely. Shorter-term exposures generated an average DREF of 4 (range 1.1 – 10). The magnitude of the dose-rate factor clearly depends on the total length of the irradiation [32].

In summarizing dose-rate effects for radiation-induced tumorigenesis, NCRP 64 recommends DREF values of 2 – 10. All subsequent examinations of the dose-rate effect, both in later NCRP publications and by other committees, rely heavily on NCRP 64 while also reviewing relevant experimental studies of the dose-rate effect published since that time. A DREF of 2 (or 1.5) is typically applied when establishing low dose-rate risk estimates. There are two reasons for using a low value of DREF. First, most committees have made the assumption that the ratio of impact between high dose and low dose radiation is the same as the ratio of impact between high dose-rate and low dose-rate radiation. That is, the dose effect factor (DEF) will be the same as the dose-rate effect factor (DREF). [Accordingly, these factors are often combined and referred to as the DDREF.] Data from the A-bomb survivors show a reduction by a factor of 2 between risk estimates generated at a dose of >

3 Gy and those generated at a dose of 1 Gy[6]. And second, lower values of the DREF are considered more appropriate for routine radiation protection because they are more conservative. For instance, in choosing to adopt a DREF of 2 rather than a larger number, BEIR V states that the higher values of DREF listed in NCRP 64 reflect situations involving continuous daily irradiation until death, but found that this "may be an unlikely circumstance for humans except as a result of natural background radiation." [2]

Residents returning home, post-accident will, in fact, be irradiated for the remainder of their lives (albeit to an ever decreasing extent) from residual ^{137}Cs in the environment. Therefore the impact of dose protraction rather than merely the true dose-rate effect *should* be taken into consideration when RHDLs and relocation triggers are considered.

As noted in NCRP 64, very long-term radiation reduces the biological consequences to a greater extent than is predicted when just using the DREF [32]. Thus, extending use of a DREF of 2 (or lower) to situations involving very long-term irradiation due to radionuclides in the environment represents a significant overestimate of the detriment. This overestimate, according to the assessment of NCRP 64, is approximately a factor of 5. BEIR VII considers a DREF of 1.5 to be most believable while "…recognizing that the choice is somewhat arbitrary and may be conservative." Overly-conservative risk estimates may have a role to play in routine radiation protection scenarios, but they become unnecessarily burdensome in the post-accident scenario when their use requires important response on the part of each individual member of the public.

Absence of data in the relevant dose-rate range: Also important regarding the analysis of dose-rate effects is the fact that none of the data examined in evaluating the DREF (or the DDREF) have been generated at dose-rates close to those of the RHDLs. Figure 3 reproduces the logarithmic dose-rate plot shown in Figure 2(b) and superimposes the dose-rates compared in each of the studies referenced by NCRP 64 in their summary table from which the DREF for tumorigenesis is obtained. Note that the lowest dose-rate examined in establishing the DREF [32] is still a factor of 100 greater than the least conservative RHDL; most of the "low" dose-rates studied are more than 1000 times greater than the RHDLs. A review of the studies cited by subsequent agencies and committees in discussions of the DREF since NCRP 64 show that this is still the case [4,6-10]: none of the laboratory studies or human exposure conditions examined address the dose-rates encountered upon returning home following a severe reactor accident[7].

One of the reasons lower dose-rate data have not been used is the fact that biological effects at lower dose-rates could not be observed. In fact, NCRP 64 found that low doses and low dose-rates lead to *increased* longevity rather than the decreased lifespan seen at higher doses and dose-rates. In addressing the apparent life lengthening at low dose-rates, the NCRP interprets this effect as reflecting "a favorable response to low grade injury leading to some degree of systemic stimulation." They go on to state that "…there appears to be little doubt that mean life

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[6] The basic assumption is that as dose-rate is lowered, a rate is eventually reached (the limiting dose-rate) where the element of time no longer matters and the effects of low dose radiation are the same as low dose-rate radiation. Confirmation that this is so, however, is based on examining dose-rates and doses that are much larger than the RHDLs. For instance, BEIR VII cites only Cornforth et al [33] in support of this assumption. This study found no difference in chromosome aberration number when two 'low' dose-rates were compared (0.105 and 0.047 cSv/min). The lack of difference was interpreted as indicating that both dose-rates must be lower than the limiting dose-rate and therefore no change in biological effect of delivery rate is seen. However, these dose-rates differ by less than a factor of 2.5 from each other, but are more than 10,000 times higher than the least conservative RHDL. It has not been demonstrated that dose-rates consistent with the RHDLs generate the same effect as dose-rates 10,000 times higher when the same total dose is given.

[7] Some data regarding repeated delivery of low doses from diagnostic radiology (sometimes referred to as 'fractionation') are used but, from the perspective of the irradiated cell, these remain high dose-rate delivery scenarios, separated in time. BEIR VII lists six studies involving low dose-rate radiation in their Table 2-1 "Dose-Response Relationships at Relatively Low Doses" [7], however none of these include dose-rates near the RHDLs (with the exception of one study that used Sellafield workers exposed to a lifetime dose of 50 mSv as a *control group* for evaluating the health impact of larger cumulative doses [34]).

span in some animal populations exposed to low level radiation throughout their lifetimes is longer than that of the unirradiated control population." [32, p 104]

Thus, the consequences of radiation exposure at dose-rates used in estimating the DDREF are quite possibly different than the consequences of the much lower dose rates typical of the RHDL. Since no data relevant to the dose-rates of the RHDLs have been evaluated in generating the DDREF, it is not possible to predict with any certainty how to modify the risk estimates generated from the A-bomb survivor population for application to the post-accident return-home scenario. Clearly, however, the uncertainty associated with applying the radiation protection risk estimates to the post-accident, long-term irradiation scenario is very large.

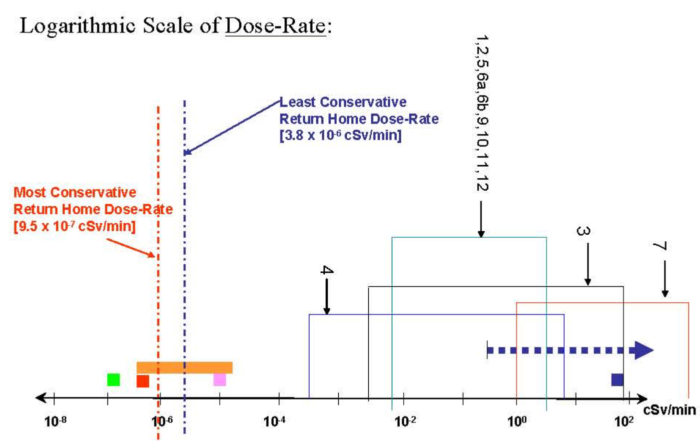

Figure 3. The logarithmic dose-rate chart shown in Figure 2(b) is displayed again here; superimposed are the dose-rate comparisons made by studies referenced by NCRP 64 in estimating dose-rate effectiveness factors (DREF) for tumorigenesis. A total of 12 datasets are cited; the dose-rates compared by each study are indicated, numbered here according to their number in Table 9.3, NCRP 64 [32]. Note that none of the comparisons examined dose rates similar to the RHDLs, or even within a factor of 100.

B.4 Uncertainty in the risk estimates:

As described above, the estimates of radiation risk are very uncertain. Uncertainty analyses of the risk per unit dose estimate generated using the LNT model for application to radiation protection have been performed by a number of committees and agencies [35,36]. Relative uncertainties about the nominal risk estimates are generally estimated to be in the range ± 200 to 400 % [7] when constraining the risk versus dose relationship to pass through (0,0). That is, the 90% confidence interval about the nominal risk estimates covers a range of risk estimates that varies by a factor of approximately 7.

Figure 4a is from NCRP 126 "Uncertainties in Fatal Cancer Risk Estimates used in Radiation Protection" (1997) and shows the ranking of seven contributions to the estimated 200-400 % uncertainty [35]. Interestingly, statistical uncertainties contribute little to the total; this is because the uncertainty estimate is based on the entire dose range including both where uncertainty is small (at high doses) and where it is much larger (at low dose).

The largest contribution to uncertainty (38%) comes from estimating the DDREF. Figure 4b shows the values of DDREF considered in the uncertainty analyses and their relative probabilities of being correct, as subjectively assessed by the NCRP [35] and the EPA [36]. Both agencies assigned a most likely value of 2.0 with the probability of larger values diminishing rapidly. No consideration is made of DREFs as large as the PFs (which are more appropriate to the return home irradiation conditions) but this would *substantially increase the uncertainty* associated with the risk estimates. [Recall the protection factor ranges from 6.8 – 12.8.]

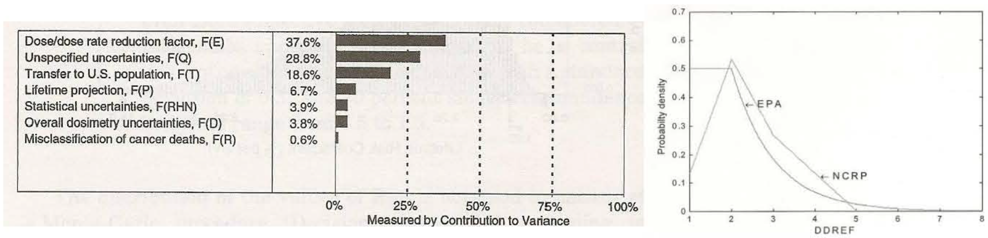

Figure 4. (a) Relative importance of various components of uncertainty to the uncertainty associated with lifetime cancer fatality risk estimates for a general population. Taken from NCRP 126 (p. 72) [35]. (b) Probability distributions for DDREF applied in the uncertainty analyses by the NCRP [35] and the EPA [36]. Taken from [36].

Assuming a very large DDREF (i.e. DDREF $\rightarrow \infty$) implies the existence of a threshold dose below which no increase in cancer fatality would be seen. The latest evaluation of the A-bomb survivor data [37] has demonstrated that a threshold dose of 0.04 Sv (4 mSv) fits the data with a statistical significance level *equal* to that observed when using the LNT model[8]. NCRP and EPA clearly state that the 'choice of dose-response model' is not considered in determination of the 200-400% uncertainty in low dose-rate risks estimates (see Figure 4a). However, if both the LNT and a threshold of 0.04 Sv (4 rem), acutely delivered, fit the data equally well, this surely implies an even greater uncertainty in the risk estimates[9].

In summary, the 200-400% uncertainty in risk estimates derived for radiation protection scenarios is likely to be a very significant underestimate of the uncertainty when applying these risk estimates to the effects of long-term, protracted exposure. This estimate of the uncertainty (i) does not consider values of DREF consistent with PFs, (ii) the data used to generate DDREFs do not include dose-rates consistent with the RHDLs, and (iii) the possibility of the dose-response relationship not following the LNT model is not considered. The uncertainty in applying radiation protection based risk estimates to the return-home scenario is likely to be at least a factor of 10 and is probably much greater.

Given this enormous uncertainty, can we really say that it makes sense to require big-impact actions on the part of the public based on doses that differ by factors of only 2 or 4? [For instance, the factor of 2 difference between the maximum allowable dose to members of the public from all man-made radiation sources for routine radiation protection (5 mSv) and the dose levels that trigger relocation of the public following an accident (10 mSv).] It cannot be in the best interests of the public to mandate extensive dose avoidance strategies, when the harm of not avoiding the elevated radiation is so unknown and when the costs for these avoidance strategies are significant and are borne by the public. It is critical that we reduce the uncertainty associated with our estimates of the harm resulting from prolonged exposure to elevated dose levels. Potential strategies for reducing this uncertainty are provided in Part C.

[8] *"Based on fitting a series of models with thresholds at the dose cutpoints in the person-year table, the best estimate of a threshold was 0.04 Gy with an upper 90% confidence bound of about 0.085 Gy. However this model did not fit significantly better than a linear model."* from Preston et al, 2007 [37].

[9] The potential existence of a dose threshold below which no excess cancer fatalities will occur has been considered as part of the SOARCA study by recalculation of the risk data assuming one of three different threshold doses. The use of the threshold models leads to a substantial reduction in the estimated risk of latent cancer fatalities resulting from exposure to radionuclides in the environment.

Part C: Recommendations

The discussion in Part B highlighted the enormous uncertainty we face when trying to predict the impact of chronic, low dose-rate radiation on human health. The SOARCA study demonstrates, however, that it is these dose conditions, almost exclusively, that we will be faced with in the unlikely event of a severe reactor accident.

As mentioned in the SOARCA documentation, little guidance as to how to estimate the likely health impact of low dose, low-dose-rate radiation exposure is provided by the national and international committees who examine available data, and the approach we borrow from radiation protection risk estimates is *not appropriate* for use in dealing with long-term exposures due to radionuclides in the environment. Therefore, while it is not the role of the NRC to dictate how the RHDLs should be set, the NRC and the industry should take a stronger position on determining the true nature of radiation-related health effects at the dose-rates anticipated following a severe nuclear power plant accident. It should be a priority.

Decades of intense effort have vastly improved our ability to predict the progression and outcome of a wide range of reactor accidents; predictions have been verified against experiments in numerous subsystems leading to greatly enhanced precision in reactor-based risk estimates. A similar deepening of our understanding has not occurred in the estimation of the impact of prolonged elevated radiation exposure on human health, the most likely consequence of a severe reactor accident. We are getting asymptotically closer to the most accurate risk estimates we'll ever generate with the A-bomb survivor population but, as discussed in Part B, this will not be good enough to tell us what we need to know in the post-accident scenario. We need to examine other irradiated populations and in particular, large populations exposed to low doses, or to low dose-rates.

C1. Diagnostic medical radiation is our largest source of radiation dose: Establish a Medical Radiation Registry for EVERYONE

Some countries operate a registry for tracking annual occupational dose for all radiation workers [38]. The existence of such a registry makes it feasible, at some time in the future, to examine health effects as a function of doses received. In most case, however, occupational doses are much *smaller* than individual medical doses [14,29] and therefore any health impact of occupational radiation may never be discernable from the potential effects of the larger medical doses. It makes more sense, therefore, to record our medical doses and to store these in a database. This we do not do.

A couple of decades ago the doses received from diagnostic radiology were relatively small and experienced by relatively few individuals. Today, however, radiological exams are used for addressing a much broader range of medical questions and are performed on a much bigger fraction of the population. More important is the fact that we've begun to make routine use of the more dose-intensive procedures of x-ray computed tomography (CT) and interventional fluoroscopy [17]. The result is that the average US resident receives as much radiation dose from diagnostic radiology procedures as from all natural background radiation sources, combined.

Thus, on a routine basis, and for a variety of reasons, we deliberately and carefully irradiate most members of the U.S. population, exposing them to a wide range of doses depending on the reason for the exam, the part of the body being imaged, and the patient's body thickness. It makes sense to maintain a registry of radiation doses for everyone irradiated. This registry would not be a "de-identified" patient radiation dose data-base (as proposed recently by the FDA as a starting point for establishing consistent exam parameters across medical institutions [39]), but a registry that allows tracking of an individual's dose over time and, ultimately, for correlation of dose with disease or health status many years later.

Initiating and maintaining a database of patient doses, if done correctly, would provide the single greatest database for low dose radiation exposures. It would also present important advantages for risk determination not available with the LSS study. First, it is unlikely the A-bomb survivor data will ever be able to provide a statistically significant determination of the risk of radiation-induced cancer as a function of dose in the low dose range. Stratification of risk estimates to ask questions about radiosensitivities based on age, gender, medical status, radiation history, etc., will also never be possible in the low dose range with the LSS. On the other hand, hundreds of millions of radiological exams are performed each year in the U.S. on people of all ages. While some radiological procedures are performed on patients with an underlying medical condition that could represent a confounding factor for any future analysis of radiation-induced health consequences, many procedures are performed in cases of acute trauma or in other situations with "no evidence of disease". Even if only a subset of the radiological exams were appropriate for long-term evaluation of the effects of radiation on health we will still quickly accumulate a sufficient sample size for the statistical power we need to answer the question: what impact do low doses of radiation have on our health? With over 350 million diagnostic radiology or nuclear medicine exams performed in the U.S. in 2006 alone [17] the statistical precision possible is very quickly greater than that with the A-bomb survivor study (<98,000 irradiated to any dose in the LSS), and we will be able to stratify the data so that we can assess the impact of low dose radiation on potentially sensitive subgroups within the population.

Second, the population we would be learning about is <u>ourselves</u>. The dataset would not contain the significant uncertainties associated with transporting the risk estimates from a Japanese population (with their substantially different organ-specific baseline cancer rates) to a general US population; this 'risk transfer' is the second largest identified contributor to uncertainty in the risk estimates (see Figure 4a). The Japanese A-bomb survivor population is also different in other ways. The population had been war-ravaged for several years and was malnourished and weakened at the time of the blasts. What impact does this have on our ability to use their cancer risk estimates and apply them to ourselves being irradiated under very different conditions? Multiple stressors on the body make us more susceptible to acute (high dose) radiation effects. Does the same apply to long term cancer induction at low doses? In that case the risk estimates derived from the A-bomb population might be far too high. But what if the effect of the bomb's devastation on societal infrastructure, and access to food, clean water, and medical services immediately after the blast led to the early deaths of the weakest of the population? This would mean that we are now measuring the effects of radiation only on the strong (a 'strong survivor' bias). In that case the risk estimates derived from the A-bomb population study would be too low. Examining the impact of diagnostic medical irradiation on ourselves would eliminate this uncertainty.

There are significant hurdles to overcome in establishing such a database; however it may represent our best opportunity for developing an understanding of low dose effects and for this reason a strong effort should be made to overcome these hurdles. Of great importance to this effort is the current move toward digitization of patient medical records. Over the next couple of years is therefore an opportune time for determining the precise parameters to capture for inclusion in the database and for interfacing with the digital patient records.

For instance, while we do not currently record the doses received by individual patients, strategies do exist for determining individual organ doses for each patient from each procedure[10] and such information could be stored in the patient's electronic record. Other data naturally included in the electronic medical record and of potential relevance to radiation response would be medications at the time of the exam, antidepressants, diet, and prior radiation history.

Analysis of a medical radiation dataset would provide a valuable supplement to the LSS data at <u>low doses</u>, essentially the only dose range of interest in the post-accident scenario. It would not, however, provide direct information about the effects of low dose-rate radiation. [Diagnostic radiology represents high dose-rate delivery, perhaps repeated (e.g. a week or a decade later) to the same or a different part of the body. Given the range of time scales relevant to human biological processes, this situation is unlikely to generate the same biological effect as the same dose spread out continually over time.]

[10] Currently we do not record the doses received by individual patients. Instead we take note of how many radiological exams of a particular type are performed each year in the country, then, on a one-time basis, we measure the dose to a 'typical' patient (using a Reference Man [20] phantom) from this exam. We then multiply the two values together for an estimate of the dose to the entire population, on average, from this particular exam. However strategies exist for determining organ doses for each patient and from each procedure.

C2. Systematic study of the health effects of residents of HBRA around the world:

For understanding the impact of chronic, low dose-rate radiation we can examine the many regions of the world whose inhabitants are already living with the dose rates represented by the RHDLs (and higher), and have been for many generations. Residents of high background radiation areas (HBRAs) do not appear to suffer adverse effects from these dose-rates, and in some cases appear to be healthier and living longer than those living in nearby control areas with lower radiation levels [15,40,41]. Such comparisons, however, often suffer from small sample size, incomplete dosimetry, and a lack of uniformity between studies that prohibits combining of the data. Many comparison studies are ecologic in design in which dosimetry data are aggregated over the entire population; this type of study is subject to several bias and correlation problems which do not arise in cohort or case/control studies where information for each individual subject (dosimetry, confounding factors, etc.) is available [42,7]. UNSCEAR has suggested that only cohort or case/control studies are suitable for quantification of radiation risk [43].

Similar problems arise with studies evaluating the impact of the Chernobyl accident or large industrial releases of radionuclides. BEIR VII lists a total of 51 ecologic studies related to the health impact following the Chernobyl accident but only 4 case-control studies [7]. Thus, although populations living with elevated radiation levels do exist, we are not yet able to use the information they can provide for the purpose of estimating the impact of elevated, chronic, low dose-rate radiation.

A concerted approach involving co-ordination of efforts across several countries and involving cohort or case-control studies could be undertaken [15,42]. Use of common study protocols and dosimetry methods will improve the robustness of the data and allow data pooling to increase statistical power. A great deal of information relevant to the issue of triggers for relocation and return-home could be obtained from a thorough and long-standing examination of these populations.

Use of such a dataset removes many of the largest sources of uncertainty associated with the risk estimates generated from the A-bomb survivor study (see Figure 4a). First, by directly examining the health impact of large populations living with a range of <u>different dose-rates</u> we can eliminate the largest source of uncertainty associated with using the LSS data for this purpose, namely the error in choosing the DDREF. Second, the uncertainty associated with translating the risk estimates from one population (Japanese) to another (US) will also substantially disappear. HBRA areas are found throughout the world so data will be generated based on similar dose-rates but in different groups with different genetic backgrounds. We will eventually be able to estimate the magnitude of differences between population groups, as well as the magnitude of the uncertainty encountered when applying risk estimates from one group to another. Further, the additional uncertainty associated with "lifetime projection" (see Fig. 4a) will also be removed since with this dataset we would examine actual fatalities, not those that are anticipated.

<u>There are several advantages to such an approach:</u>

1. Dose-rates in HBRA span a considerable range. It would be possible to examine health consequences of a full range of dose-rates including those we are now using to trigger relocation and return-home, as well as dose-rates considerably higher. In addition to examining health of the general population in a HBRA, it would be possible to investigate impact on sub-groups within the population. Analysis of the A-bomb survivor data indicate that women are somewhat more sensitive than men to radiation-induced cancers, and that children are substantially more sensitive than adults with the sensitivity changing remarkably with age at time of exposure [44]. The deeper question is: are the same variations in radiosensitivity expected at low dose-rates and to small doses?
2. The wide range of genetic make-up represented by those living in HBRA around the world provides the opportunity to examine the range of genetic susceptibilities to radiation-induced effects by comparing each group living in an HBRA with its own control group (a similar population but living with lower background levels).
3. There may be particular diets, medications, or even lifestyles that affect radiation sensitivity. For instance, Lemon et al have shown that including a mix of anti-oxidants in the diet of mice results both in increased

longevity and reduced DNA damage (a 6-fold reduction in chromosome aberrations) following a single acute dose of radiation [45]. The neurotransmitter serotonin has been shown to impact the ability of irradiated cells to transmit information about the radiation event to neighboring cells [46]. Would such chemicals as antidepressants, for example, affect the body's response to low dose-rate radiation? And similarly, are there mitigators (medicine) we should be taking when living in an elevated background? Much effort is underway to develop mitigators to redress the effects of high dose/high dose-rate radiation resulting from terrorist activities. Similar efforts could be undertaken to determine if mitigators are useful or necessary when living in slightly elevated backgrounds levels.

4. Given the dearth of data on irradiated humans, animal data are often used to extend our understanding of radiation-induced health effects. However, not all animals show similar responses (as a function of time or of dose) as humans and thus it is difficult to know when and how to extrapolate from animal to human data [47]. Examination of the indigenous animal life, comparing those in the control areas with those in the HBRA, may give us the opportunity to study the magnitude of species-to-species or strain-to-strain differences in response to radiation, perhaps shedding light on the degree to which we can rely on animal data to inform our understanding of human radiobiology.

5. Much of the increase in radiation dose in a HBRA comes externally from elevated gamma-emitting radionuclides in the soil, similar to the irradiation route anticipated with the post-accident return-home scenario. In some areas, however, substantial increases in internal radionuclides contribute to elevated dose through food and water consumption. How different are the effects of higher organ doses when the dose is delivered via external gamma rays than when the elevated dose results from eating food elevated in ^{137}Cs or other radionuclides? In still other HBRAs, the increased radiation is due to higher elevations where contributions to dose from cosmic radiations are increased [40]. These radiations have higher LET but the impact of low dose-rate high LET radiation is not currently well understood.

6. The number of people living in some HBRAs is very large, potentially providing the necessary statistical power to generate statistically significant estimates of the differences in health impact of elevated background radiation. Over 20 million people live in Mexico City (elevation 2240 m) where the cosmic radiation dose is more than double the world average [41]. More than 350,000 residents of the HBRA in Kerala India are currently being studied [42] via interviews to examine factors associated with lifestyle, medications, etc., and dosimetry has been performed in and around over 70,000 homes. Over 125,000 residents of the HBRA in Guangdong Province in China have been under study since 1987 [48].

7. Is an increase in our natural cancer rate the *only* thing we need to worry about? Data arising from analyses of the A-bomb survivors suggest that other health effects may be elevated (although the data are insufficient for detailed investigation). Data from animal studies suggest the relative causes of death as a function of age change after exposure to radiation [49].

C3. Summary:

Both strategies proposed here for developing an improved understanding of the health impact of low dose or low dose-rate radiation will require a very significant effort to generate the data we need to develop our understanding of the real consequences of a severe reactor accident. However there is considerable on-going effort to take advantage of. For instance, experience with life-long tracking of radiation dose is available with radiation registries used to track occupational radiation doses [38], and of course, extensive high quality analysis of long-term health effects from radiation exposure as a function of age and gender has been performed for decades by the Radiation Effects Research Foundation in their analysis of the participants in the Life Span Study [27,37]. This existing experience would represent a valuable starting point, for instance, for a registry of individual patient doses.

The information captured would be of great use to society, and not only for dealing with the consequences of a severe reactor accident. A more precise understanding of low dose radiation effects would be useful in diagnostic medicine (the exponential growth of CT in the US has far outpaced scientific knowledge of the effects of CT-relevant doses on human health; are these doses safe?), space travel (elevated low dose rate exposure, particularly in deep space), and weapons after-effects (far more people will be exposed to low-dose prompt radiation or chronic radiation from fall-out following a nuclear weapons attack than will suffer the acute radiation syndrome).

In all these scenarios we currently rely on the LSS to inform our understanding of the potential health impact of low doses and/or low dose-rate radiation and, as described in Part B, our understanding is very minimal.

For several reasons it is important to begin this effort now. First, the A-bomb survivor study has shown there is a 20-30 year latent period between acute exposure and a cancer diagnosis so it will be many years before we can begin examining health-related consequences of radiation doses. Second, since the move toward digital patient records is underway now, we will lose the opportunity to define which parameters to capture and to store for subsequent analysis of radiation-related impact if we do not use this time to weigh-in on what these records should contain. And third, we need the information that such a database and analysis would provide before an accident occurs. As noted by the EPA, while it would be possible to lower the RHDLs after an accident if it were justified, it would probably not be possible to increase them [1].

As discussed in Part B, most of the data we currently use to evaluate health impact are high dose, high dose-rate data. We need to begin focusing on that dose regime relevant to the post-accident scenario. Emphasis should be on providing best estimate information. One way to get started would be to repeat the Expert Solicitation study (NUREG/CR 6555) [5], asking different questions.

The risk estimates from NUREG 6555 are used, after modification by a DDREF of 2.0, in the quantification of latent cancer fatality risk in the SOARCA study. Questions in NUREG 6555 were put to 13 experts and deal with cancer incidence and cancer death rates for a given radiation dose. For all but one case involving exposure to low LET radiation, the experts were asked to provide their estimate of the risk of fatal cancer resulting from high dose, high dose-rate radiation (1 Gy delivered over 1 minute). Not surprisingly, all experts made use of existing A-bomb-based risk estimates and variation between predictions of the experts was small [5].

Of greater use would be expert estimates of the effects of low dose-rate and low dose gamma radiation, i.e. conditions identical to those we will encounter upon returning home following a reactor accident. [An abundance of recent data relevant to these questions comes from experimental studies of life-long irradiation of small animals, and radiobiological examination of low dose and low dose-rate radiation on cells and tissues in the laboratory (e.g. impact of doses below 5 mGy [50-52]). The field of radiobiology has undergone rapid changes in the last decade with advancements in biological assays and interrogation methods that make it possible to address biological responses at lower doses than possible in decades past.] Expert opinion should be solicited regarding the risks associated with long-term, protracted radiation exposures of the public. Such risk estimates will be far more useful in generating a 'state-of-the-art' estimate of the consequences of elevated radionuclides in the environment.

The industry has done an excellent job of increasing the depth of understanding of reactor technology, accident progression, behavior of fuel and thermal-hydraulic systems under various conditions, radionuclide dispersal, meteorological modeling, particulate deposition patterns as a function of weather, and evacuation planning. Our ability to model and predict the dose that someone will receive from a particular accident initiation sequence that leads to radionuclides in the environment is considerable. However that is where our knowledge stops; we know very little about the impact of this dose (more particularly the dose-rate) on our health. In other words, the real consequence of a reactor accident leading to elevated radiation levels is the impact on human health and we do not know what this impact will be.

We evaluate accident risk in units of "reactor years". As we anticipate increasing our dependence on nuclear power-generated electricity then it is only prudent to develop a thorough understanding of the consequences of the increased radiation exposure we can expect in the unlikely event of a severe reactor accident, an unlikely event whose likelihood increases with every new reactor brought on line or every license renewed.

Appendix

Glossary of Acronyms:

ALARA	As Low as Reasonably Achievable
BEIR	Biological Effects of Ionizing Radiation (US National Academies of Sciences)
DEF	Dose Effectiveness Factor
DDREF	Dose and Dose-Rate Effectiveness Factor
DREF	Dose-Rate Effectiveness Factor
DS02	Dosimetry Study 2002
EPA	Environmental Protection Agency (US)
HBRA	High Background Radiation Area
ICRP	International Commission on Radiological Protection
LET	Linear Energy Transfer
LNT	Linear No Threshold (model of radiation response with dose)
LSS	Life Span Study (A-bomb survivor dataset)
NCRP	National Commission on Radiological Protection (US)
PF	Protraction Factor
RHDL	Return home dose limits
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation

Units of Dose (and effective dose equivalent):

500 mrem	= .005 Sv = 5 mSv
2 rem	= 0.02 Sv
100 rad	= 1 Gy
100 rem	= 1 Sv
1 Gy	= 1 Sv (for low LET radiation from gamma emitters in the environment)

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Peer Review Comment Resolution Report

April 30 - December 3, 2010

The purpose of this report is to provide the Peer Review Committee with the State-of-the-Art Reactor Consequence Analysis (SOARCA) team's resolution of each peer review comment related to the best estimate analysis and received between April 30 and December 3, 2010. In some cases, previous comments (received before the April 2010 draft peer review report) were included to provide a complete understanding of the entire comment. This report is organized by individual peer reviewers (see the Contents below). The source document and its associated page number are identified for each comment. Comments are extracted directly from the source documents with no changes. The SOARCA report sections that are identified in the resolutions refer to the revised reports that will be provided to the Peer Review Committee before the final meeting on the best estimate analysis.

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Note: Mr. Roger Kowieski indicated in the April 30, 2010 peer review draft report and the December 3, 2010, peer review memorandum that all of his comments had already been adequately addressed.

KEN CANAVAN

April 30, 2010 SOARCA Peer Review Draft Report

1. **Page 10**

Comment:
There is a possibility that certain accident sequences, while not-dominant, may have increase risk in terms of increased consequences. While these sequences may not dominant the risk, in terms of either frequency and/or consequence, they could be contributors. Collections of several lower order sequences could have higher consequence than SOARCA evaluated and could also contribute. While SOARCA did indeed capture the most likely sequences and did accurately capture the consequence from these sequences.

Resolution:

As explained in Section 2.1, "Approach," of the Main Report, the State-of-the-Art Reactor Consequence Analyses (SOARCA) team considered and selected accident scenarios (sequence groups, rather than individual sequences) based on both likelihood and potential consequences. Core-damage sequences from previous staff and licensee probabilistic risk assessments (PRAs) were identified and binned into core-damage groups. A core-damage group consists of core-damage sequences that have similar timing for important severe accident phenomena and similar containment or engineered safety feature operability. The *groups* (not individual sequences) were screened according to their approximate core-damage frequencies to identify the most significant groups. Since core-damage groups, i.e., scenarios, were considered, many individual lower-order sequences would be captured in the aggregation into groups.

The SOARCA team used selection criteria based on core damage frequency (based on the latest NRC SPAR models) and initiating events involving containment bypass or leading to an early failure of the containment with potential consequences. Scenarios (sequence groups) with a frequency at or above 10^{-6} per reactor-year were screened in, as well as scenarios with a frequency at or above 10^{-7} per reactor year for scenarios that are known to have the potential for higher consequences (e.g., containment bypass scenarios such as steam generator tube rupture and interfacing system loss-of-coolant accident (ISLOCA) initiators). Please note that these criteria were not rigidly or inflexibly applied. In addition, some scenarios were included even though they did not explicitly meet our screening criteria. For example, candidate

SOARCA sequences were examined with radiological release timing in mind (both the timing of core damage and the timing of containment failure) and considered for inclusion since a major impact on both early and latent cancer fatality risks is derived from the timing of the offsite release. The Peach Bottom short-term SBO scenario was included, even though it did not satisfy our selection criteria, because it has a more prompt radiological release and a slightly larger release.

The selected sequence groups for SOARCA were shown to be important in recent and past probabilistic risk assessments, particularly for the well-studied pilot plants of Peach Bottom Atomic Power Station (Peach Bottom) and Surry Power Station (Surry). The scenarios selected for SOARCA were compared to NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," to corroborate that important scenarios were captured. The selected scenarios are also representative of broad classes of transients. The selection of station blackout (SBO) events in SOARCA ensures that we have covered that broader class of transients involving a loss-of-heat removal and further, by including a short-term station blackout (STSBO) we have reasonably bounded that class of accidents (which could include other events such as loss of service water or loss of component cooling water but which develop more slowly). Also, for the PWR, the station blackout also includes, in part, the effect of a small loss of coolant by considering reactor coolant pump seal leakage. In addition, by the selection of station blackout sequences for analysis, we also include the effects of loss-of-containment heat removal (fan coolers) and loss-of-containment spray systems (which are all electrically powered) to remove airborne radionuclides.

The purpose of SOARCA was to do a detailed best estimate consequence analysis. Pursuing a Level 3 PRA which would capture consequences from additional lower order sequences would necessitate introducing conservative assumptions in order to complete analyses and would have served a different purpose. The SOARCA study is not a traditional risk study which is intended to quantify total risk from all sequences, but instead focuses on quantifying the consequences of important severe accident scenarios.

In addition, the most recent Level-3 PRA-type information that we have, while somewhat simplistic and approximate, indicates that SOARCA is likely to have considered accident scenarios that account for the majority of risk at both the Peach Bottom and Surry plants. The Severe Accident Mitigation Alternatives (SAMA) analysis submitted with the license renewal application (2001) for Peach Bottom indicated that station blackout scenarios accounted for the majority of offsite risk (Reference: Peach Bottom Atomic Power Station, License Renewal Application, Environmental Report, Appendix G, available at: http://www.nrc.gov/reactors/operating/licensing/renewal/applications/peach-bottom/peach_bottom-envg.pdf). The SAMA analysis for the Surry license renewal application (2001) indicated that SGTR and ISLOCA accounted for the majority of the off-site population dose risk (Reference: chapter 5 of NUREG-1437, Supplement 6 (2002), "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Regarding Surry Power Station, Units 1 and 2 – Final Report," available at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1437/supplement6/2671bpart2.pdf).

2. **Page 10**

Comment:
The SOARCA analysis and report is developed by applying a method to two specific plants Surry and Peach Bottom. The application of the methods to two specific plants has both positive and negative aspects. The positive aspects are that with plant specific information, plant specific conclusions can be drawn and can be based on the specific design features, maintenance and operation practices at that particular site. The downside to this approach is that not all the plant specific features, both those features that reduce consequences as well as those that might increase consequences, are represented in the two plants chosen. As such, some conclusions are likely applicable to that site only and the results may not be typical.

Resolution:
The SOARCA analyses are indeed plant specific evaluations, and should not be used out of context. The team is aware that caution should be taken in projecting these results to other similar plants; however, insights from these analyses are generally expected to be useful for other plants. The appropriateness of applying individual insights to other plants should be evaluated on a case-by-case basis. This point will be included in the conclusion section of the main report.

3. **Page C-2**

Comment:

The objectives of the SOARCA project appear in several locations. In some of these locations the wording is slightly different. It is recommended that a single list of goals and objectives be developed and used consistently.

Resolution:

The document is being revised to refer to objectives in a consistent manner. Preliminarily, the objectives in the current draft are:

The overall objective of the State-of-the-Art Reactor Consequence Analysis (SOARCA) project is to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. Corresponding and supporting objectives are as follows:

- Incorporate the significant plant improvements and updates not reflected in earlier assessments including system improvements, training and emergency procedures, offsite emergency response, and recent security-related enhancements described in Title 10, Section 50.54(hh) of the *Code of Federal Regulations* (10 CFR 50.54(hh)) as well as plant updates in the form of power uprates and higher core burnup.

- Incorporate state-of-the-art integrated modeling of severe accident behavior that includes the insights of some 25 years of research into severe accident phenomenology and radiation health effects.

- Evaluate the potential benefits of recent security-related mitigation improvements in preventing core damage and reducing an offsite release should one occur.

- Enable NRC to communicate severe-accident-related aspects of nuclear safety to stakeholders including Federal, State, and local authorities; licensees; and the general public.

- Update quantification of offsite consequences found in earlier NRC publications such as NUREG/CR-2239, "Technical Guidance for Siting Criteria Development."

4. Page C-2

Comment:
The abstracts in the reports are not used as effectively as they could be. Formal abstracts will be the location where the authors can summarize their findings, results and conclusions, and methods. These are important aspects of the report and it is recommended that they be fully developed.

Resolution:
The abstracts in the report will be reevaluated with the reviewer's suggestions in mind, and revised as necessary to ensure a high-level summary of findings, results and conclusions is captured.

5. **Page C-2**

Comment:
In addition, Executive Summaries are also not well utilized. Additional care could make them more effective.

Resolution:
The SOARCA team is in the process of revising the Executive Summary.

6. **Page C-2**

Comment:
Seismic research issues and the treatment of seismic issues have the general impression that their contribution would be a foregone conclusion. The area of seismic sequence conclusions reached are valid but the uncertainties associated with the occurrence of large seismic events as well as the consequences of such an event are high. This should be acknowledged in the report. In addition, on-going seismic research efforts should also be addressed or acknowledged.

Resolution:
Additional text was added to Appendix B Section 3.5, "Surry Seismic PRA (SPRA) Study," comparing the recent Surry Seismic Probabilistic Risk Assessment Pilot Study with the events that occur in SOARCA. The Surry SPRA study produced a total CDF which is comparable to the CDF of the SOARCA external events (on the order of 2 x 10-5/yr). The dominant scenario (comprising 50% of the total CDF) was identified as a loss of service water (LOSW). While

consideration of such an event was not considered in SOARCA, the long-term SBO (LTSBO) scenario may qualitatively serve as a surrogate for the LOSW event in certain respects. In both instances, the plant is undergoing a loss of heat removal transient with potential reactor coolant pump seal leakage. In addition, the Surry SPRA study assumes the emergency condensate storage tank (ECST) and the fire protection water tanks are not available due to their low capability to withstand seismic loading. If indeed these tanks are assumed to fail catastrophically this would result in immediate loss of auxiliary feedwater (AFW) unless other sources of water for the AFW system can be aligned. It is clear that the worst-case LOSW scenario as identified in the Surry SPRA study combined with the loss of the ECST would still be bounded by the unmitigated short-term SBO scenario, since that scenario credits no primary side injection or auxiliary feedwater. The SOARCA team plans on discussing this comparison at our next Peer Review Committee meeting.

NRC is currently conducting ongoing seismic research, some of it in cooperation with EPRI under the NRC-EPRI Memorandum of Understanding. However, as this was not the focus of the SOARCA study, this ongoing research is not highlighted in the SOARCA reports.

7. **Page C-2**

Comment:
Several factors that anecdotally support the conclusions of lower consequences in SOARCA include changes to the physical plant and procedures. Some of these changes include the Station Blackout Rule, the ATWS Rule, development of Emergency Operation Procedures (EOPs) and Abnormal Operating Procedures (AOPs), plant specific simulators, severe accident management guides (SAMGs), the maintenance rule, and overall improved performance. These should be mentioned in the report.

Resolution:
The Station Blackout Rule, ATWS Rule, and Maintenance Rule are mentioned in the Main Report in Section 2.5, "Generic Factors," as factors that contribute to the decrease of risk estimates for nuclear power plants over the years. EOPs are discussed in Section 2.2, "Scenarios Initiated by Internal Events," of the Main Report and SAMGs and 10 CFR 50.54(hh) mitigation measures are discussed in Section 3.1, "Site-specific Mitigation Strategies," of the Main Report. EOPs and SAMGs (as well as other 10 CFR 50.54(hh) improvements for

mitigated scenarios), and how they are incorporated into the analysis, are now discussed at the beginning of Section 2, “Accident Scenario Development,” in Appendices A & B.

CLÉMENT

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8. **Page 13**

Comment:

Concerning the accident progression for Surry, one of the most important results of the analysis is that a creep rupture of the hot leg nozzle occurs before induced failures in other locations of the RCS and before failure of the lower head of the reactor pressure vessel. It is also considered that the rupture of the hot leg nozzle results in a large break. This has important consequences for what happens next. First, the depressurization of the RCS allows injection of water by the accumulators that delays the progression of the accident. Secondly, this avoids any high pressure melt ejection. In addition to this base case, scenarios with thermally-induced steam generator tube rupture were considered. Although the base case scenario is credible and corresponds to the best-estimate philosophy of SOARCA, uncertainties on different failure modes and locations must be taken into account.

Related Comment (p. C-3)**:**

The answer to comment #1 is not satisfactory, as no variability was introduced in the timing of RPV lower head failure. This could be done, as discussed at the last meeting, through a sensitivity study on parameters governing the relocation of corium to the lower head.

Resolution:

While hot leg rupture is predicted deterministically prior to thermally induced-steam generator tube rupture (TI-SGTR), an induced tube rupture sequence was selected as a variant of the Surry Station Blackout analysis. The most significant competing challenges occur between hot leg (HL) failure versus a TI-SGTR; the parameters that govern the timing of HL creep rupture relative to the TI-SGTR were examined. Section 5.3.3, “Uncertainties in the Failure of the Thermally-Induced Steam Generator Tube vs. Hot Leg,” in Appendix B examines the sensitivity of the timing of hot leg failure to the TI-SGTR. It was the conclusion of the current MELCOR analyses and the extensive number of analyses supporting NUREG/CR-6995, “SCDAP/RELAP5 Thermal-Hydraulic Evaluations of the Potential for Containment Bypass During Extended Station Blackout Severe Accident Sequences in a Westinghouse Four-Loop PWR,” that failure of these piping components (i.e., the HL and pressurizer surge line piping and SG tubes) is the most likely failure locations. The piping failures are encountered shortly after the time when the oxidation power peaks. Melting and relocation of the core fuel to the RV lower head are seen to occur long after the time of the piping failures.

Relative to the Dr. Clément's comment, the potential for other failure modes and locations must be taken into account. Based on considerable research including analysis of accidents, experiments, system code calculations, and CFD simulations, the most likely locations for RCS failure include: (1) the HL nozzle; (2) the steam generator tubes; (3) the pressurizer surge line; and (4) the lower head[1]. As discussed above, analyses of failures at the HL nozzle and the steam generator tubes were considered. Due to the presence of reactor coolant pump seal leakage, surge line failure was very unlikely because the seal leakage (and/or seal failures) prevented PORV cycling during core damage[2]. The final RCS failure location would be the lower head prior to other substantial RCS failures (i.e., if unmitigated, all scenarios eventually lead to lower head failure). For example, in the MELCOR sensitivity calculation where hot leg failure was prevented (see Appendix B Section 5.3.3), vessel failure was predicted to occur at 5.2 hours, nearly 2 hours after the TI-SGTR. During the 2 hours following the TI-SGTR, the hot leg creep rupture index exceeded the failure criteria by four orders of magnitude. A realistic combination of events or uncertainties has not been identified that would decrease or disable hot leg or whole loop natural circulation and the associated RCS structural failures (i.e., HL or TI-SGTR) for such a long duration. The basis for this conclusion also considered the results of NUREG/CR-6995, which includes the results of extensive code parameter variations, equipment response uncertainties, and operator actions. While conceptually possible, it was considered incredible for the scope of the present study (e.g., the same reason that an alpha mode failure was not considered).

9. **Page 14**

Comment:

The analysis shows that hydrogen combustion by jet ignition becomes possible after the hot leg rupture. Bounding cases are given for adiabatic, isochoric, complete combustion (AICC) and

[1] In addition to these locations, recent research by industry and NRC suggests that thermal failure of the instrument tubes leading to release into the containment is another potential location. Although this may have some significance in the B&W once-through steam generator design, it would occur much later than HL failure or TI-SGTR in a U-tube Westinghouse steam generator and therefore is not relevant.

[2] If the pressurizer surge line failed, the subsequent response is similar to a HL failure. Prior to the inclusion of RCP seal leakage and/or seal failure (i.e., circa 2000), the pressurizer safety valves continued to cycle during core damage, which preferentially induced hot gas flow from the core into the surge line. This led to failure of the surge line as dominant failure location.

detonation. It would be interesting to see if we are far or not from the σ criterion for flame acceleration and the λ criterion for detonation in order to evaluate.

Related Comment (p. C-3):
This comment refers to the presentation made by KC Wager at the last meetings. It is stated that a jet ignition is likely after hot leg creep rupture failure. Bounding cases are given for AICC and detonation. It would be interesting to see if we are far or not from the criterion for flame acceleration and the criterion for detonation. Those are given in the following document: "W. Breitung et al., "OECD State-of-the-Art Report on Flame Acceleration and Deflagration-to-Detonation Transition in Nuclear Safety," NEA/CSNI/R (2000)7, August 2000."

Resolution:
To address uncertainties in the STSBO for hydrogen combustion, the SOARCA team updated Section 5.2.3, "Uncertainty in the Hydrogen Combustion in the Mitigated STSBO," in Appendix B which reports the results of sensitivity analyses undertaken in response to peer reviewers' concerns. Based on the sensitivity analyses, the SOARCA team concludes that the best-estimate response reported in Section 5.2.2, "Mitigated Short-Term Station Blackout," is a reasonable representation of the source term, and any uncertainties in the best-estimate response should not have a material effect on the best-estimate results.

Section 5.2.3 describes the analysis of the potential for jet ignition following hot leg failure and vessel failure. In addition, a sensitivity study was performed that precluded any combustion until bulk detonation conditions were achieved (i.e., for the case with emergency spray mitigation). The adiabatic isochoric complete combustion (AICC) model for deflagration and the Chapman-Jouguet (CJ) model for detonation were used to assess the resultant pressure challenge (i.e., the results showed the peak pressure would exceed the best-estimate containment failure pressure). Additional 2-dimensional shock calculations were also performed. The reviewer recommends additional research to further evaluate the potential for flame acceleration from deflagration to detonation. At the time of hot leg failure, a hydrogen rich jet exits the hot leg. The bulk hydrogen values calculated at this time were relatively low and the steam content was high. Consequently, it was concluded that there was insufficient hydrogen to support the propagation of a detonation. Later, after emergency spray operation and further hydrogen production, there is high hydrogen and low steam content in the containment but the airborne fission products were knocked down by the same sprays that decreased the steam inerting potential. Future work may be appropriate to apply the referenced methodology to assess the likelihood of deflagration to detonation transition. But such future work is outside the scope of the current SOARCA study. As noted above, based on the sensitivity analyses, the SOARCA team concludes that the best-estimate response reported for Surry in Appendix B is a reasonable representation of the source term, and any uncertainties in the best-estimate response should not have a material effect on the best-estimate results.

10. Page 14

Comment:
No transport of gaseous iodine in the RCS is considered although this was experimentally evidenced. There is also no treatment of gas iodine chemistry in the containment. The Project made a sensitivity study to cope with this modeling lack: gaseous iodine concentrations observed in the Phebus FPT-1 experiment were added to the containment inventory. Phebus was an NRC funded project using MELCOR code conducted at Sandia National Labs. As the calculated iodine releases are already high, this addition does not make a big difference. It should however not be forgotten that this would probably not be true for other sequences with lower releases. Also, it is expected that gaseous iodine releases due to gas phase chemistry phenomena in the containment could last for a longer time that the 48 hours considered in the studies.

Resolution:
The SOARCA team acknowledges that gaseous iodine remains a source term issue, especially with respect to long term containment performance issues after the comparatively much larger airborne radioactivity has settled from the atmosphere. We consider the mechanistic modeling treatment for gaseous iodine behavior to be a technology still under development with important international research programs underway to determine the dynamic behavior of iodine chemistry with respect to paints, wetted surfaces, buffered and unbuffered water pools undergoing radiolysis and gas phase chemistry. We believe that the base case treatment under our best practices recommendation is sufficient for the best-estimate effects addressed in SOARCA, and we plan to investigate parameterization of the gaseous iodine fraction of total iodine releases in the uncertainty analysis for the project.

11. Page 15

Comment:

Progress has been made in the recent years in the knowledge of accident progression and source term evaluation. Not all the outcomes have been incorporated in MELCOR models and advances in knowledge are still ongoing. It should be valuable, when a MELCOR version incorporating significant new features becomes available, to benchmark the present SOARCA results with this new version for some selected sequences.

Resolution:
There would be value in reassessing such studies as new significant source term information becomes available. Nonetheless, the SOARCA team does feel the SOARCA analyses are consistent with the present state-of-the-art.

12. Page C-3

Comment:
Synthesis report pp. 11-12 – Some words could be added about the uncertainties on accident progression. Not only the weather conditions and their consequences will be considered in the uncertainty analyses.

Resolution:
Revised Section 1.9, "Uncertainty Analysis," now includes additional discussion of uncertainties in accident progression. In addition, although uncertainties will be addressed in the uncertainty analysis in an integrated fashion, many sensitivity studies have been conducted since the March 2010 Peer Review Committee meeting that investigated uncertainties identified in the accident progression analysis. The results of many of these sensitivity analyses are included in the SOARCA reports, and some of these will be discussed at our next Peer Review Committee meeting.

13. Page C-3

Comment:
The answer to comment #4 by Clement clarifies the use of CDF as screening criteria. It would be valuable to add this text in the final report as well as parts of answer to comment #5 and to comment #58 by Leaver.

Resolution:

Chapter 2, “Accident Scenario Selection,” of the Main Report is a detailed presentation of the accident scenario selection process, and emphasizes the key points in the answers referenced in this comment.

14. Page C-3

Comment:
The treatment of comment #3 gives a correct answer, showing low consequences on RN releases. It should however be good, in the future, to consider a distribution of initial defects in the SG tubes, obtained from inspections' feedback experience.

Resolution:
A considerable amount of work has been done by the NRC analyzing the potential for thermally induced steam generator tube rupture (TI-SGTR) (e.g., NUREG/CR-6995). SOARCA incorporated the findings from these studies to include the potential for TI-SGTR. Two cases were considered: a single tube rupture and two tubes. The failures were assumed to occur near the steam generator inlet plenum tube sheet where the high temperature stream enters the tube bundle. The one and two tube TI-SGTR cases showed interesting and divergent effects of enhancing oxidation and providing additional core cooling, respectively. As the reviewer notes, additional realism could be introduced by reviewing data from plant inspections to examine the location and magnitudes of defects. This data could be cross-correlated against the CFD work done by the NRC (i.e., NUREG-1788, “CFD Analysis of Full-Scale Steam Generator Inlet Plenum Mixing During a PWR Severe Accident”) against the likelihood of those tubes receiving the highest temperature flow stream from the inlet plenum. In effect, it was assumed that the most vulnerable defect(s) was exposed to the highest temperature gas stream entering the SG. Since the high temperature stream cools rapidly as it flows through the steam generator, the timing of the failure for other locations would only decrease the timing between the subsequent hot leg failure. It was believed that the SOARCA approach was conservative in this respect but consistent with the current state-of-the-art. Additionally, separate multi-year work is underway, in a different NRC project, to assess the likelihood of TI-SGTR, considering updated flaw distributions and material changes. Future analyses can benefit from additional realism by considering inspection experience.

JEFF GABOR

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15. Page 16

Comment:
Due to the primarily deterministic approach taken, great care must be taken in communicating these results in any context that include a discussion of risk to the public. The project and associated documentation details a more realistic assessment of the potential consequences associated with operating nuclear reactors for the accident progression scenarios evaluated and portrays a more up-to-date understanding of the key accident phenomena.

Resolution:
The SOARCA team has a separate explicit task from the Commission to ensure that SOARCA is communicated effectively to the public. The importance of accurately portraying the results of this study and its implications on risk is understood by the SOARCA team. This concern regarding communication results has been expressed by other peer reviewers too, and is addressed in other responses as well. The SOARCA study is not a traditional risk study which is intended to quantify total risk from all sequences, but instead focuses on quantifying the consequences of important severe accident scenarios. The study does, however, provide quantification of individual health risk for the accident scenarios selected.

16. Page 17

Comment:
As the SOARCA project did not evaluate a full spectrum of scenarios, great care must be taken in the communication of these results. While potentially representative, these results are plant-specific, limited in scope, and do not fully characterize plant risk.

Related Comment (December 3 Memo, p. 2):
Care should be taken in characterizing SOARCA in the context of a “risk” study. Since final documents have not been provided to the Peer Review Team, it is not confirmed that this has been addressed.

Resolution:

This aspect of the study has been a subject of ongoing discussion. Unlike PRA's such as the NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," study or contemporary PRA's, whose objective is to address a fuller risk profile, SOARCA is intended to be an assessment that identifies and quantifies consequences (and scenario specific risk) from the more important accident scenarios. We believe that the detailed examination of the specific scenarios selected in SOARCA provide a realistic view of potential consequences. See also response to Mr. Canavan's Comments 1 and 2.

17. Page 17

Comment:
Lower Head Penetration Failure – Comments were provided as to the omission of lower head penetration failure as a possible vessel failure mode. The SOARCA analysis did not include these failure mechanisms based on the fact that the majority of BWR accident sequences are assumed to result in the RPV being depressurized prior to core relocation into the lower head. It is acknowledged that the likelihood of these failure mechanisms is reduced at lower RPV pressures.

Related Comment (p. C-4):
[MELCOR Best Modeling Practices] Section 3.1.1.5 - I would recommend a little more explanation on why penetration failure as a mode of vessel breach has been ignored. This needs additional justification.

Related Comment (December 3, 2010 Peer Review Memo, p. 2)**:**
BWR lower head penetration failures are not represented. Clear justification has not been provided.

Resolution:
BWR penetration failure is considered to be a complex failure mode. At the time of material relocation to the lower head, both the lower head and the guide tubes are filled with water. It is assumed that this water must first be evaporated before significant thermal attack occurs. If guide tube failure occurs prior to lower head failure, there is still no clear pathway for movement of core materials out of the vessel region, owing to the presence of drive tube mechanisms and considerable structural materials residing in the drive tube below the vessel head. Concerning the lower head drain plug, we feel that this penetration will likely be plugged by lower melting

point control blade materials and will not be predominantly affected by subsequently relocating core materials. Owing to the complexity of modeling this detail, we elected to model only the vessel head structure itself. Without experimental information to guide such complex modeling, the approach taken in the SOARCA project is considered commensurate with the current state of knowledge. As noted in Section 4.3.1, “Peach Bottom MELCOR Model,” of the Main Report, a penetration failure model was not used because the timing differences between gross lower head failure and penetration failure with the available penetration model are not significant to the overall accident progression (i.e., minutes difference). Also, Sandia lower head failure tests showed gross creep rupture of the lower head was measured to be the most likely mechanism for vessel failure (Chu, T.Y., et al., NUREG/CR-5582, SAND98-2047: Lower Head Failure Experiments and Analyses. 1998, Sandia National Laboratories: Albuquerque, NM.).

18. Page 18

Comment:
Hydrogen ignition in SBO - comments were provided to identify the source for hydrogen ignition in the station blackout sequences. Section 5.1.3 of the Surry Accident Analysis was updated to include a more thorough discussion of ignition sources. Hot gases exiting the reactor vessel upon hot leg creep rupture and at the time of lower head failure were shown to have sufficient energy to ignite the hydrogen. An additional investigation was performed to study hydrogen combustion upon mitigation using containment sprays. Prior to spray recovery the containment atmosphere can be inerted by the steam present; however, as the steam fraction is reduced from spray actuation, small burns are shown to occur. My review comment addressed a possible delay in hydrogen ignition upon spray actuation and Section 5.1.3 was revised to include this sensitivity.

Resolution:
It is the SOARCA team’s interpretation of this comment that the reviewer is satisfied with the revised additional analysis of the spray actuation and hydrogen ignition.

19. Page 18

Comment:
SRV failing in the open position- the SOARCA analysis identified SRV sticking open during core heat-up as the dominant mechanism for causing RPV depressurization. Competing phenomena includes the heat-up and potential failure of the Main Steam Line nozzle. As a result of my

comments, Section 5.6 of the Peach Bottom Integrated Analysis includes a substantial analysis of the uncertainty associated with the SRV failure mode. Cases were included assuming an early failure of the SRV, a failure but with only ½ of the relief area, and a case without SRV failure but with subsequent creep failure of the main steam line nozzle. These sensitivity cases provide valuable insights and show that the highest release of iodine to the environment is associated with the MSL creep failure case. Where it is understood that the SOARCA development team believes that SRV failure case represents the best-estimate, it would be useful to show the consequence impact due to the MSL failure case. In addition, the impact of the hot gas on the potential for Drywell head failure resulting from the MSL failure was not considered. The sensitivity of the results to this failure mode are further evidence that focus on the analysis and reporting of individual accident progression scenarios can be misleading. This is why a fully risk-informed approach to the presentation of consequence information is preferable.

Related Comment (December 3 Memo, p.2)**:**
Potential for Main Steam Line failure not justified. Where there has been a sensitivity case provided for MSL creep failure, the impact on consequences has not been calculated. There needs to be a strong technical case developed for why there will not be a MSL creep failure. Even with a lower expected probability for MSL creep failure, higher consequences could make this important.

Resolution:
In order to address Dr. Gabor's comment on how the SOARCA team believes the SRV failure case is a best-estimate, the SOARCA team reanalyzed the SRV cases identifying two conditions that would lead to the MSL creep rupture. Results of these MELCOR calculations are described in Section 5.5.2, "Extreme Variations in Failure Criteria," of Appendix A. These MELCOR calculations determined that the following conditions are necessary for MSL creep rupture: 1) stochastic SRV failure being ignored, and over temperature failure of an SRV not being considered until valve stem temperature reached a very high value (1,175K); or 2) thermal failure occurs at the best-estimate temperature criterion, but the resulting open fraction of the SRV is very small (10% or less). These conditions are judged to be extremely unlikely given the current understanding of SRV failure, and thus were not considered representative of the best-estimate analysis. Therefore, no consequence calculations were conducted. Many sensitivity

studies were performed exploring a wide range of means by which MSL failure would occur. Please note that this issue will also be looked at as part of the uncertainty analysis.

20. Page C-4

Comment:
I believe Dr. Henry previously identified this, but it would be good to include a discussion of the differences between BWR and PWR core. This could be added to [MELCOR Best Modeling Practices] Section 3.1.3 and simply explain the differences (channel boxes, etc.) and provide some discussion of their impact.

Resolution:
Section 4.1, "Reactor Vessel and Coolant System," and 4.2, "In-vessel Structures and Reactor Core," of Appendix A provide information on the BWR reactor and core configuration and Section 4.1, "Vessel and Reactor Coolant System," of Appendix B provides information on the PWR. From these two appendices a reader can identify the differences between a BWR and a PWR core. More information will be provided in these sections to discuss how unique features of both types of core designs are modeled in MELCOR.

21. Page C-4

Comment:
[MELCOR Best Modeling Practices] Section 3.1.1.5 – I would also recommend some discussion of structures in the lower plenum (instrument tubes, CRD tubes, etc.) and an indication of what their impact would be. This is another area where differences between BWR and PWR could be highlighted.

Resolution:
As noted in the response to Comment 20, additional information will be provided in Sections 4.1, "Reactor Vessel and Coolant System," and 4.2, "In-vessel Structures and Reactor Core," of Appendix A to discuss the manner in which unique design features of the BWR (versus PWR) lower plenum are modeled in MELCOR.

22. Page C-4

Comment:

[MELCOR Best Modeling Practices] Section 3.1.1.6 – I would recommend some discussion of the impact of structures in the cavity area on debris spreading and cooling.

Resolution:
The SOARCA team is not aware of any structures within the BWR reactor pedestal that would significantly impede the flow of debris, except for the pedestal wall itself. An open doorway from the pedestal to the main drywell floor restricts (or directs) flow from the pedestal to the drywell floor to a small fraction of the azimuthal circumference of the floor. But this restriction is accounted for, as described in Section 4.5, "Behavior of Ex-Vessel Drywell Floor Debris," of Appendix A.

23. Page C-4

Comment:
[MELCOR Best Modeling Practices] Section 3.1.1.7 – I recommend an explanation of why they assume a PWR valve will fail at the cumulative failure probability of 50% and a BWR valve at 90%.

Resolution:
The PWR analysis selected 50% failure probability from the beginning of the SOARCA analysis to represent median failure conditions. A 90% failure probability was used in the initial BWR calculations to represent a "high confidence" level for an event that was perceived to be a 'benevolent failure,' That is, a failure that would lead to a more delayed and smaller source term. These different modeling approaches developed independent of each other, and the inconsistency was recognized later as a consequence of questions raised by Peer Review Committee. When the SOARCA analysis was revised to address these (and other) peer review comments, the differences in failure criteria narrowed, but were also found to be unimportant to the results, as explained below.

The approach used to model stochastic failure of an SRV to reclose in the BWR analysis was replaced by a more 'best estimate' approach based on early peer review comments. The revised criterion for stochastic SRV failure was defined based on the "expected value" for the number of cycles a valve would experience at the time of failure. 'Expected value' is calculated as 1/failure-rate. If one translates this approach to a cumulative probability at the time of failure, the value corresponds to a 63% confidence level for BWRs, which is closer to, but still different from, the (assumed) 50% probability used in the PWR analysis. The calculated number of cycles experienced by primary and secondary coolant system relief and safety valves in the PWR is much less than the number corresponding to the median (50%) failure probability. Therefore, stochastic failure never occurs in the PWR calculations. Confidence in this observation would only increase if the failure condition were shifted from the median failure probability to the probability corresponding to the "expected value" (63%). Therefore, an adjustment to the PWR model was not deemed necessary. A footnote will be added to Appendix B to explain that the exact cumulative failure probability chosen did not matter for Surry.

24. Page C-4

Comment:
[MELCOR Best Modeling Practices] Section 3.1.3.1 – This section needs to discuss Drywell shell failure. Section 4.3 even points here for such a discussion.

Resolution:
Additional discussion of the relationship between lateral debris mobility and drywell shell melt-through as a containment failure mechanism have been added in Appendix A Sections 4.5, "Behavior of Ex-vessel Drywell Floor Debris," and 4.6, "Containment Failure Model."

25. Page C-4

Comment:
[MELCOR Best Modeling Practices] Section 4.2 – For completeness, DCH in a BWR should be discussed and reasons for it being a low threat included.

Resolution:
MELCOR Best Practices Section 4.2, "Direct Containment Heating (DCH)," includes the discussion of DCH in PWRs. DCH in BWRs has not been as extensively investigated as with PWRs. However, similar to PWRs, opportunities for depressurization in the reactor vessel prior to that DCH event exist, such as operation of the safety relief valve system (SRV), or sticking or seizure of the SRVs. Focused studies within the SOARCA project conclude that SRV seizure open is very likely to occur well before any potential direct containment heating event. A short discussion on DCH in BWRs will be added to the MELCOR Best Practices document.

26. Page C-4

Comment:

The end [Appendix A] Section 4.5 raises “drywell liner melt-through” as one of the containment failure modes considered. It points the reader to Section 4.4, however, there is no discussion on liner melt-through. I recommend that there is a brief statement on what liner melt-through is and what the assumed criteria for failure is. It is clearly stated that water will prevent it, but no details are ever provided on what the failure model/criteria is. It might also be helpful to indicate the assumed area of failure and maybe a discussion of the release of pathway associated with failure mechanism.

Resolution:

New text has been added to Sections 4.5, “Behavior of Ex-Drywell Floor Debris,” and 4.6, “Containment Failure Model,” of Appendix A, as noted in the response to Comment 24.

27. Page C-4

Comment:

[Appendix A] Section 5.2 – LTSBO discussion: I recommend a statement on the assumed operator action to vent the containment. It only shows up on the figure with no discussion. PCPL is closer to 60 psia, so venting at 40 psia needs to be explained and perhaps a description of the “possible” release pathway. I just think that this action needs to be called out in the text somewhere.

Resolution:

The principal basis for the containment venting criteria used in the original calculation of the mitigated long-term station blackout (LTSBO) (reviewed by the Peer Review Committee) was comments received from the licensee during a verbal walk-through of LTSBO mitigation, which was part of a site visit in 2007. Based on this comment (and others) received from the Peer Review Committee, the criteria were re-examined and a new (replacement) calculation has been performed, which is identical to the earlier one except for the assumed criteria for opening/closing the hard-pipe containment vent line. The new calculation assumes the hard-pipe vent path was opened at 45 psig and reclosed at 25 psig. These values were selected based on a review of plant-specific procedures for containment pressure control (Peach Bottom procedure T-102), but also taking into consideration isolation setpoints for RCIC, which are sensitive to containment thermodynamic conditions. Most important in this regard is the setpoint for high turbine exhaust pressure. The Peak Containment Pressure Limit (PCPL) suggested in procedure T-102 is 60 psig. However, a high turbine exhaust pressure isolation signal for RCIC would be received at a pressure of 50 psig. Since RCIC is the only operating coolant injection system available in this scenario, we assumed operators would open the containment vent path at 45 psig, thereby averting RCIC isolation.

Results of the revised calculation are the same as those obtained in the earlier calculation except for the containment pressure response. Discussion has been incorporated into the updated version of Appendix A.

December 3, 2010 Peer Review Memorandum

28. Page 2

Comment:
Most aspects of SOARCA represent a plant-specific evaluation and should not be extrapolated to other plants.

Resolution:
SOARCA analyses are indeed plant specific evaluations, and should not be used out of context. The team is aware that caution must be taken in projecting these results to other similar plants; however, insights from these analyses may be useful for other plants. The appropriateness of individual insights for other plants must be evaluated on a case-by-case basis.

29. Page 2

Comment:
For mitigated Surry SBO, the initiation of containment sprays may result in an early containment failure due to hydrogen combustion. A clear justification has not been provided for why this will not be a significant contributor to off-site consequences.

Related Comments (received before April 2010):
Mitigated short term SBO: why are there H2 burns? Is there a criterion for ignition when there is no power? Is nodalization controlling? What would be the impact of delaying the burns due to inadequate ignition?

Hydrogen burn (deflagration) was discussed, but there was no discussion of hydrogen detonation. Has this been evaluated to be below the CDF defined? In this reviewer's experience, hydrogen detonation, depending on their size and location, can cause large leakage or breach of containment

Resolution:
A number of hydrogen combustion and detonation studies were conducted as part of the SOARCA analysis. Only delayed hydrogen combustion was shown to be a threat to containment integrity. In this case the radionuclide concentration in the containment environment was too low to result in a significant environmental release.

Section 5.2.3, "Uncertainties in the Hydrogen Combustion in the Mitigated Short-Term Station Blackout," of Appendix B was updated and examines uncertainties in the time of combustion and the impact of hydrogen detonation. No additional consequence calculation was performed on these calculations because the source term was limited to noble gases.

See the resolution to Comment 9 for further discussion.

30. Page 2

Comment:
Uncertainty analysis needs to be performed.

Resolution:
An uncertainty analysis is underway and draft results are expected to be available by mid 2012. We will be discussing the uncertainty analysis in more detail in a future Peer Review Committee meeting.

ROBERT HENRY

April 30, 2010 SOARCA Peer Review Draft Report

31. Page 21

Comment:

Throughout the report, there are numerous places where the "Objective" of the SOARCA assessment is defined. These all relate to the best estimate nature of the evaluations but the statements are not identical. For something as important as the objective of the study, the wording should be agreed upon and either be repeated exactly, or referenced, (to another part of the study), every place where this needs to be discussed. From my perspective, the important aspects of SOARCA are as follows:

- The central estimate/calculation of every aspect of the study is focused on the best estimate which is an appropriate focus for a state-of-the-art examination.
- This study is supported and directed by the Nuclear Regulatory Commission so it should be clearly stated that this study is specific to the U.S. fleet of commercial nuclear power plants. Clearly these are representative of a BWR and a PWR, with each having one of the important containment types used in the U.S.
- The studies include several plant specific features associated with the RCS and containment design, EOPs, SAMGs, etc. Hence, this shows the important influence of several plant specific features that have been included as operator actions, etc. that are taken during the accident progression.

Therefore, I suggest that the objective statement for the SOARCA be something like what is in the Abstract of the Summary document, but with some additional text. My suggestion is as follows:

The primary objective of the SOARCA project is to provide a best estimate evaluation of the likely consequences of important severe accident events at reactor sites in the U.S. civilian nuclear power reactor fleet. To accomplish this objective the SOARCA project has applied integrated modeling of accident progression and off site consequences using both state-of-the-art computational analysis tools to two previously analyzed reactor sites (Peach Bottom and Surry). To meet the state-of-the-art objective, the analysis tools utilized best modeling practices drawn from the collective wisdom of the severe accident analysis community. Equally important, the analyses for both of the reactor sites also represented the implemented procedures in the main control room and elsewhere, that are relevant to the response for the *important accident conditions related to highly unlikely, but possible radiological releases.*

Resolution:

The document is being revised to refer to objectives in a consistent manner. Preliminarily, the objectives in the current draft are:

The overall objective of the State-of-the-Art Reactor Consequence Analysis (SOARCA) project is to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. Corresponding and supporting objectives are as follows:

- Incorporate the significant plant improvements and updates not reflected in earlier assessments including system improvements, training and emergency procedures, offsite emergency response, and recent security-related enhancements described in Title 10, Section 50.54(hh) of the *Code of Federal Regulations* (10 CFR 50.54(hh)) as well as plant updates in the form of power uprates and higher core burnup.

- Incorporate state-of-the-art integrated modeling of severe accident behavior that includes the insights of some 25 years of research into severe accident phenomenology and radiation health effects.

- Evaluate the potential benefits of recent security-related mitigation improvements in preventing core damage and reducing an offsite release should one occur.

- Enable NRC to communicate severe-accident-related aspects of nuclear safety to stakeholders including Federal, State, and local authorities; licensees; and the general public.

- Update quantification of offsite consequences found in earlier NRC publications such as NUREG/CR-2239, “Technical Guidance for Siting Criteria Development.”

32. Page 22

Comment:

The inclusion of a MELCOR “best practices” document is a very important feature of the SOARCA evaluation. It defines the manner in which the accident progression for both BWRs and PWRs was evaluated as part of these central estimate calculations and also provides some of the features that are to be explored through the upcoming uncertainty analyses. In that regard, it is necessary that the best practices document describes the manner in which the evaluations were performed. It is important that the review committee reviews and comments on the controlled features associated with the MELCOR calculations.

Resolution:
The SOARCA team agrees that the MELCOR 'best practices' is a very important feature of the present study. The team will present information to the peer reviewers at a planned forthcoming meeting and information will be included in the Peach Bottom and Surry Reports. Please note that the MELCOR Best Practices document is expected to be available by late 2012.

33. Page 22

Comment:
In the current draft, there is a good description on the manner in which "breakout" of molten zirconium through a thin layer of oxidized cladding is evaluated in the MELCOR code for these analyses. This relocation of metallic zirc is an important feature associated with the overall melt progression. In addition, there is an extensive discussion of the dominant chemical states of the fission products and how these are evaluated in terms of the release rates from the oxide fuel and into the high temperature gas space of the Reactor Coolant System (RCS). There is also an extensive discussion on the modeling approach for cesium molybdate release rates for the fuel. In the current version, much of this appears to be written as part of the PWR description. However, these features are common to all of the BWRs and PWRs in the U.S. commercial fleet and should be clearly described as such in the write-up.

Resolution:
Additional text was added to Appendix A to include the relevant discussion for BWRs. In addition, other features have been included to document the manner in which the central estimates were evaluated. See response to Comment 24 for a specific example. The MELCOR Best Practices report will also be revised accordingly and is expected to be available by late 2012.

34. Page 22

Comment:

The release fractions of the dominant chemical states provides the manner in which the fission products from the fuel become airborne in the core region. The transport of these fission products from the core, thorough the RCS and into containment, as well as their deposition in these regions is determined by the aerosol model. Typically, the aerosol densities within the reactor coolant system can be in excess of 100 grams per cubic meter, which is a very dense aerosol. Hence, the deposition within the RCS can be quite large and the manner in which this is calculated needs to be documented as part of the “best practices”. I suggest that the benchmarks of the aerosol model with experiments such as the large scale ABCOVE tests, the DENONA test, etc., where available, be included in this “best practices” document. This is important to capture since the aerosol transport and deposition model is that feature of the calculation that determines the extent of airborne fission products in the containment that could be released to the environment. It is particularly key that this discussion be included, along with the benchmarks that are relevant to the aerosol densities typically encountered in the RCS and containment, to be assured that indeed a central estimate is justified.

Resolution:

Currently, it is planned that analyses demonstrating code model assessment and validation will be documented and maintained in the MELCOR Code Manual Volume 3. The last published version of this report was for MELCOR version 1.8.5 and it is being updated for code Version 2.1. Essential validation exercises for the most part are not strongly dependent on the code version, as MELCOR models are relatively mature at this point. For example, validation analyses were last performed on the ABCOVE and Marvekin tests with code Version 1.8.2 but will be revisited with MELCOR 2.1 in the near future. The importance of validating the aerosol mechanics, transport and deposition modeling is well appreciated, and considerable validation in this area presently exists for MELCOR, including the VANAM experiments (similar to DEMONA and also performed in the Battelle Model Containment facility), LACE, CSE, and smaller scale facilities such as FALCON and AHMED, where aerosol agglomeration, hygroscopic effects and deposition, settling and spray scrubbing models are assessed and validated. The following table summarizes the status of code validation tests, including aerosol tests, for various versions of MELCOR and the plans for future documentation. Some discussion concerning code version and validation status follows.

Assessment\Code Version	1.8.1 (1991)	1.8.2 (1992)	1.8.3 (1994)	1.8.4 (1995)	1.8.5 (1996)	1.8.6 (2005)	2.1 (2011-2012)	MELCOR 1.8.5 Volume 3	CONTAIN Parity	Phebus Synthesis	IBRAE Assessments	Separate Assessments[3]	MELCOR 2.1 Volume 3
FLECHT-SEASET	x						x				x	x	x
LACE-LA4							x	x	x				x
CSE Spray Experiments			x		x				x			x	x
Marvekin	x										x	x	
DEMONA		x										x	x
ABCOVE Tests		x					x					x	x
NUPEC Mixing Tests					x		x	x	x				x
Ahmed Hydroscopic Tests					x	x	x		x				x
VANAM-M3 (ISP37)					x		x	x	x				x
IET DCH Experiments					x		x		x				
HDR-V44					x		x		x				x
HDR E-11							x					x	x
FALCON Tests							x				x		x
NTS Hydrogen Burn Tests					x		x		x				x
BWR Mk-III Vent Clearing Tests				x	x		x		x				x
GE Level Swell Tests			x				x		x			x	x
PNL Ice Condenser test	x				x				x			x	
RTF Iodine Tests (ISP41)					x				x				x
BESTHSY (ISP-38)							x				x		x
PBF-SFD1-4							x				x		x
LOFT-FP2	x					x	x				x	x	x
TMI-2					x	x	x	x				x	x
Phebus B9+					x			x					x
Phebus FPT-1					x	x	x	x		x		x	x
CORA 13	x				x			x					x
ACRR MP-1/MP-2		x										x	
ACRR DF-4		x										x	
RASPLAV Salt Tests							x				x		
RAS MEI Tests							x				x		x
NEPTUNE Experiment							x				x		x
SURC MCCI							x				x		x
HI/VI FP Tests					x		x			x			x
VERCORS 1-6 & HT/RT FP Tests					x		x			x			x

[3] "Separate Assessments" means that there is a stand-alone report (SAND or NUREG report) that documents the work.

Aerosol mechanics for nonhygroscopic aerosols is modeled using the MAEROS code (analogous to the NAUA code) where good verification of aerosol agglomeration physics and gravitational depletion was demonstrated in early versions of MELCOR based on MARVEKIN, ABCOVE and LACE testing. MELCOR Version 1.8.5 introduced extensions to treat hygroscopic aerosol effects where good validation against the VANAM M3 test (similar to DEMONA) as well as the AHMED experiments was demonstrated. The CSE A9 test was used to validate the containment spray scrubbing modeling in MELCOR on code Version 1.8.5 in the CONTAIN-MELCOR parity assessment study. The CONTAIN-MELCOR parity study introduced numerous other containment behavior assessments including the NUPEC mixing tests, the Nevada Test Site hydrogen burn tests, and the IET DCH containment heating experiments. Fission product release from fuel, including MOX and High Burnup were assessed against ORNL HI/VI tests and against more recent VERCORS experiments and documented in the Phebus Synthesis report using MELCOR Version 1.8.5. In Version 1.8.5 fission product release models were adjusted using sensitivity coefficient over-rides to the Version 1.8.5 models. These were formalized as code options and defaults for code Version 1.8.6. MELCOR Version 1.8.6 also introduced expanded modeling detail for core melt progression processes, including molten pool convection treatments. These extensions provided improved prediction of the TMI-2 accident, some of which are still currently under assessment. The Phebus FPT-1 test stands as the most comprehensive integral assessment of core damage progression, hydrogen generation, fission product release and RCS deposition and containment natural depletion processes. This test provides good assessment of key deposition behavior in the reactor RCS and for containment depletion. MELCOR Version 2.1 is largely identical to Version 1.8.6 with respect to model pedigree, the main difference being conversion of the source code to FORTRAN 95. Other code assessments for code Version 2.1 have been performed by IBRAE during the code Version 2.1 conversion process as indicated in the table. The MELCOR 2.1 code manual Volume 3 will largely compile the amassed collection of code assessment problems. The MELCOR Best Practices report is intended to document the specific manner that the latest code version is being applied to specific applications such as SOARCA plant models. It is our strong contention that code Version 1.8.6 largely embodies the collective state of the art with respect to MELCOR severe accident progression models with specific plant application treatments documented in the current MELCOR Best Practices report (the SOARCA revision of the Best Practices document is expected to be available in late 2012).

The validation of code modeling, including aerosol deposition, will be discussed in more detail at the next Peer Review Committee meeting and documentation of that information will be incorporated into the SOARCA revision of the MELCOR Best Practices report.

35. Page 22

Comment:

The accident progression within the core region from the intact fuel rods with “breakout” of molten zirconium that drains to the lower core region, eventual relocation of the molten debris from the core to the lower plenum and the controlling heat transfer to the RPV lower head need to be described. With the differences in geometry between the PWR and the BWR, as well as some potential differences between the U.S. commercial fleet PWR designs, for example upflow versus downflow core bypass, this should be described in the “best practices” document since this will be consulted, evaluated and referenced in future studies. Furthermore, only through an understanding of this core melt progression and relocation to the lower plenum can the features that should be investigated through uncertainty analyses be clearly defined.

The general public is well aware of the severe core damage accident that occurred in the Three Mile Island Unit 2 reactor. Any integral thermal-hydraulic model that is used to assess the timing for the onset of core damage, the release of fission products from the core, the extent of hydrogen generated in the core degradation, the transport of molten core debris to the lower plenum, etc. needs to be benchmarked with this accident. This benchmark evaluation needs to be either part of the SOARCA documentation or, at the minimum, referenced extensively in the other SOARCA reports.

Resolution:

Much of this has been documented in the MELCOR Code Manual Volume 3 document and will be referenced in the current reports to allow the reader further access to the information. Central values of key MELCOR parameters are derived from the assessments in this document (MELCOR Volume 3). The MELCOR Best Practices report includes an overview of the numerous benchmark activities undertaken under the MELCOR project, and makes reference to the MELCOR code manual set, which includes Volume 3 (SAND2001-0929P) with MELCOR Version 1.8.5 which is a collection of benchmark and validations analyses that include TMI-2. The molten Zr “breakout” parameter, as well as other MELCOR COR-Package parameters affecting core melt progression such as rod-collapse criteria and effective UO_2/ZrO_2 liquefaction behavior is discussed in the MELCOR Best Practices. These parameters affect the amount of hydrogen generated, peak core-region temperatures and core melt progression timing and behavior, including melt relocation rate and freezing behavior. The central values for these important parameters are derived principally from comparison historically to CORA, PBF, ACRR DF-4 and LOFT FP-4 experiments, and more recently to the Phebus experiments, principally FPT-1 and the TMI-2 benchmark analysis. Assessment/validation against CORA, LOFT-FP2, Phebus FPT-1 (soon to add FPT-3), and TMI-2 are included in the MELCOR Volume 3.

36. Page 23

Comment:
Evaluating accident progression of severe accidents in BWRs and PWRs involves the physical modeling of many complicated and interrelated processes. Given that these are both complicated and interrelated means that there are numerous uncertainties that need to be considered in developing best estimate analyses. These uncertainties need to be identified in the documentation and their influence on the conclusions of the study must be included in the final assessment.

Resolution:
Sensitivity studies have been conducted since the March 2010 and October 2010 Peer Review Committee meetings that investigate uncertainties identified in the best-estimate analysis. The results of these sensitivity analyses have been included in the best-estimate reports, and these will be discussed at our next Peer Review Committee meeting. An uncertainty analysis is underway and we will be discussing this in more detail in a future Peer Review Committee meeting.

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37. Page 3

Comment:
The SOARCA best estimate analyses, and the ongoing uncertainty analyses, are needed by the NRC to remain current with the developing state-of-the-art for severe accident research and to effectively regulate the new designs that are to be submitted for certification. To ensure that the MELCOR accident sequence calculations represent best estimate evaluations, it is important to have the various models in the code benchmarked to experimentally observe results.

Resolution:

Please see the resolution to Comment 38.

38. Page 3

Comment:

Provide entire code benchmarks

- TMI-2
- Phebus

Individual models

- NRC/EPRI/Westinghouse Steam Generator Natural Circulation Tests
- MPA Stuttgart Hot Leg Creep Test
- SNL Lower Head Creep
- HDR E11.2 and T31.5 Tests
- ABCOVE Large Scale Aerosol Tests
- DEMONA Large Scale Aerosol Tests

Those denoted in the above list were specifically identified in my previous comments. It is important that the benchmarks utilize the same logic/methodology in establishing the nodalization that is used in the SOARCA plant calculations. Furthermore, the parameters varied in the uncertainty study should also be used in the benchmarks and the numerical values should be close to the central estimates for the "uncertainty variations."

The committee has asked for this background material in various ways and has yet to receive any such information. Computer code benchmarks that qualify the code calculations are the necessary foundation to the study.

Resolution:

It is currently planned that code benchmarks such as the ones cited by Dr. Henry, will be added to and maintained in the MELCOR code manual Volume 3 on Assessments and Benchmarks as established validation benchmarks for the MELCOR code. A summary of code benchmarks performed to date is provided in a table in the response to Comment 34. In comparison, the “Best Practices” document is intended to be a living document (with specific dated versions) that describes what is deemed to be our current best modeling practices regarding model inputs that may be different from the default values in the last published MELCOR User Guide Manual, or that reflect user inputs that are unique to the system being modeled. An example is the user input reflecting latest current understanding of cesium speciation, specifically the treatment for Cs_2MoO_4, derived from the Phebus experiments, that differs from MELCOR default values in the last published version of the MELCOR User Guide. In some cases, benchmark analyses are presented in the Best Practices document to document the rationale for the current best practice input values; however, in time, it is intended that such best practice input values will be made default values or documented code user options and described in subsequent versions of the MELCOR User Guide. Many of the recommended benchmark problems can be found in the SAND2001-0929P, MELCOR Computer Code Manuals Version 1.8.5 Volume 3. Chapter 13 of this code manual volume is a comparison to TMI-2. For the ABCOVE comparisons refer to SAND-94-2166, "MELCOR 1.8.2 assessment: Aerosol experiments ABCOVE AB5, AB6, AB7, and LACE LA2." Separate assessments against the Westinghouse 1/7th scale tests have been performed and serve as the basis for current MELCOR counter-current natural circulation treatment, as described in the draft version of the Best Practices document. Current default values used in the MELCOR Larson Miller treatment for lower head creep rupture are taken from the Sandia and OECD Lower Head Failure experiments. It will be verified that these are documented in the MELCOR User Guide. ABCOVE experiments are to be added to the code benchmarks archived in the code manual Volume 3. The VANAM M3 test, very similar to the DEMONA experiments, assessing aerosol depletion for hygroscopic aerosol, is already included in the Volume 3 documentation. Test HDR E11.2 has been assessed under a separate study focused on a comparative evaluation of CONTAIN and MELCOR, along with many other containment experiments including the IET tests, NUPEC Mixing Tests, ISP-41 Iodine Chemistry, HDR V-44, AHMED, Lace LA-4, CSE A9, the NTS H2 Burn Experiments and the GE Suppression Pool Venting Experiments. These containment assessment experiments are planned for inclusion in the MELCOR Volume 3 documentation.

DAVID LEAVER

April 30, 2010 SOARCA Peer Review Draft Report

39. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:
My comment 8a. asked why no mitigated STSBO sequences (i.e., STSBO sequences with 10CFR50.54(hh) measures considered) were included for Peach Bottom. The reason given in comment resolution was that no mitigated STSBO was addressed since the outcome would be the same as mitigated LTSBO (i.e., no core damage). This is not correct. According to the SOARCA analysis, STSBO will cause core damage even if the 50.54 (hh) portable injection is started at 3.5 hours (as it was in the LTSBO). A correct comment resolution should be documented. Also, if it is decided not to include a mitigated STSBO for Peach Bottom, the reason should be provided in the SOARCA report.

Related Comment (December 3, 2010 Memo, p. 7):
Appendix B – Comment 84: The reason given for the my comment on the need to provide a mitigated STSBO for Peach Bottom was that no mitigated STSBO was addressed since the outcome would be the same as mitigated LTSBO (i.e., no core damage). This is not correct. According to the SOARCA analysis, STSBO will cause core damage even if the 50.54 (hh) portable injection is started at 3.5 hours (as it was in the LTSBO). A correct comment resolution should be documented. Also, if it is decided not to include a mitigated STSBO for Peach Bottom, the reason should be provided in the SOARCA report.

Resolution:
The Peach Bottom STSBO proceeds too rapidly to allow for alignment of portable injection; therefore, a mitigated STSBO is not included. It has been judged that alignment of portable injection pumps aimed at mitigating the STSBO cannot be achieved in time to prevent core damage. Therefore, a mitigated STSBO sequence utilizing 10 CFR 50.54 (hh) measures is not performed. However, since RCIC is an existing, in-place, safety system and since RCIC blackstart procedures are in place, we have performed a STSBO calculation assuming RCIC blackstart at one hour. This calculation has been included in Appendix A, Section 5.3, "Short-Term Station Blackout with RCIC Blackstart." The blackstart of RCIC mitigated the scenario by substantially delaying the onset of core damage. This additional time would allow opportunities for further mitigation (to arrest core heatup) using the portable diesel-driven pumps. Also

blackrun of RCIC would allow for further mitigation. An explanatory discussion will be added to Appendix A.

40. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:
There are some places where the operator mitigation strategy assumed (or not) in SOARCA is questionable. One example is comment 1 above. Another is for Surry STSBO where there are reasons why the operator might install portable vessel injection as opposed to installing portable containment spray (operator will not necessarily know if and when lower head failure occurs, so he/she may opt to inject in the hope of preventing lower head failure; he/she may also opt to inject in the hope of minimizing the chance of induced SGTR; finally, the portable injection pump may be able to be installed sooner than 3.5 hours (3.5 hours was assumed for the Surry LTSBO) which would prompt the operator to go down this path as opposed to containment spray. It is suggested that a table and/or text be included in the report that presents the mitigation strategies (50.54(hh)) and the basis for the particular strategy and timing assumed so as to qualitatively strengthen the justification for mitigation success.

Resolution:
The analysis of the mitigative actions was carefully reviewed with plant operations personnel numerous times, especially in the cited STSBO scenario. Through the plant's severe accident management guidelines, there are a number of alternative mitigation options. In discussions with the plant, the range of mitigative actions were identified that would be applicable to the specific scenario and the known conditions. The implementation of the specific actions would be coordinated through the technical support center as these actions fall outside the normal emergency operating procedures. As pointed out by Dr. Leaver, this step requires human evaluation in conditions with little if any instrumentation and perhaps difficult conditions from a seismic event.

Due to the judged difficult accessibility conditions and the lack of alternating current and direct current power, the transient would be more rapid and not allow recovery in-vessel. As pointed out, a TI-SGTR could also be a concern. An extensive mitigative study was not performed but rather an assessment of one action, containment sprays using the emergency Godwin pump. The plant confirmed the preference of this action and its implementation, given the assumption that no external independent pump could be implemented before 8 hours. In a recent fact check comment response from the plant, however, they estimated that it would take approximately 2 to 2.5 hours to ready an external independent pump under normal conditions (based on table-top exercises and timed testing); this will be referenced in the Surry report.

It was judged that, due to difficult accessibility conditions, vessel failure would have occurred before an emergency depressurization using the pressurizer, PORVs, emergency injection into the vessel, or emergency feedwater could be established. Hence, this late mitigative strategy was selected as the best option. The benefits and drawbacks of this strategy were investigated through many MELCOR sensitivity calculations. In particular, the operation of the sprays could lead to large deflagrations and/or detonations. However, the sprays also cooled the ex-vessel debris and knocked down the airborne fission products. Consequently, the mitigative strategy showed net benefits.

In SOARCA, the mitigative selection exercise was done with assistance of plant personnel. The selected actions were based on the best option for the specified conditions. The relative likelihood of selecting the less likely actions was not assessed. However, it should be noted that the failure to successfully implement a mitigative action was examined in the unmitigated case. A probabilistic evaluation would consider a range of strategies and evaluate their relative likelihood and consequences. However, for the best-estimate purpose of SOARCA, only the most likely option was examined.

Text has been added to the report which discusses NRC staff holding table-top exercises with the licensee to strengthen the basis and timing for particular strategies.

41. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:
Appendix A, Figure 85, 20 mile risk (STSBO with RCIC blackstart) is higher than Figure 87, 20 mile risk (STSBO with no RCIC). At other distances it is the other way around (which is intuitively the way it should be. i.e., with RCIC blackstart, the risks are lower). This should be explained in the text.

Resolution:
There was an error in the calculations, and these figures and text have all been replaced because the source terms were recalculated and, as a result, the consequences were

recalculated. The trend that triggered this concern is no longer present in the current results; the 20-mile risk now follows the other distances, i.e., the risk for STSBO with no RCIC is higher than the STSBO with RCIC blackstart (see Appendix A Section 7.3.2, "Short-Term Station Blackout with Reactor Core Isolation Cooling Blackstart").

42. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:
Page 68 of the Summary report still says that risks are calculated to 100 miles.

Resolution:
This has been corrected in the Main Report. Section 5.8 Risk Metrics Reported states 0 to 50 miles for the mean population weighted risks.

43. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:
It is suggested that the fifth bullet in the conclusions on page xxix of the Executive Summary be generalized to apply to all sequences that were screened as opposed to just bypass sequences. For example: "Scenarios which are lower frequency than the scenarios which survived the screening criteria would not pose a higher latent cancer fatality risk than the scenarios which survived the criteria since the higher conditional risk is offset by the lower frequency."

Resolution:
The SOARCA team agrees with the comment in general. This principle is cited in connection with our discussion of the appropriateness of the screening criteria, which were also supplemented by other considerations such as a comparison with important scenarios in past studies such as NUREG-1150. Further, the results of our analyses have confirmed the principle for the other scenarios analyzed. Nonetheless, generalization of the principle to cover all probabilistic considerations, while appealing, is beyond the scope of the study.

44. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:
Suggest changing middle sentence of large paragraph on page 10 of Summary report as follows: "While it is judged, on the basis of the procedures and training, that these measures are expected to be effective, a limitation of this approach is that a comprehensive human reliability

assessment has not been performed to quantify the probabilities of plant personnel succeeding in implementing these measures." QED

Resolution:
The Main Report Section 1.7, "Mitigated and Unmitigated Cases" currently has this statement to discuss the lack of a comprehensive human reliability analysis in SOARCA:

"A limitation of this approach is that a comprehensive human reliability assessment has not been performed to quantify the probabilities of plant personnel succeeding in implementing these measures. However, NRC has issued 10 CFR 50.54(hh) requiring plant licensees to possess the equipment, develop the strategies, and have trained personnel to implement these mitigative measures. The 10 CFR 50.54(hh) measures are the result of a major effort by industry and NRC in the 2004–2008 timeframe to develop means to mitigate events involving loss of large areas of the plant due to fire and explosions.

The assessment of mitigation measures has continued to receive attention since our initial assessment conducted with plant staff. NRC staff performed follow-up site visits in June and August 2010 to explicitly address RCIC blackstart and run for STSBO and manual operation of TD-AFW. The site visits included a review of RCIC blackstart and run procedures, additional tabletop exercises to address conservatism in the assumed PWR STSBO timeline, and plant walkdowns of equipment areas and detailed reviews of procedures. SOARCA staff concluded, following these site visits, a greater likelihood of implementing mitigation." In addition, the recent post-Fukushima inspections at Peach Bottom and Surry provide some additional information with respect to the two plants' abilities to implement SAMGs and B.5.b measures (the inspection reports are available at: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/follow-up-rpts.html). The SOARCA team is currently evaluating the information in these inspection reports for any relevant insights.

Note that a sensitivity calculation is provided (the "unmitigated" scenarios), to show what the consequences would be without mitigation, which bounds the spectrum of possible consequences with regard to assigning success probabilities to the 10 CFR 50.54(hh) mitigation actions. See further discussion in response to Comment 49 and Mr. Mrowca's Comments 60-61.

45. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:

Summary report, page 22: fourth bullet, frequency range is 1E-7 to 5E-7, not 8E-7.

Resolution:

This has been corrected in the report.

46. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:

Appendix A, Section 5.5 Loss of Vital AC Bus E-12 is titled "Mitigated Response", but it is actually unmitigated per Section 3.3.3 and 3.3.4.

Resolution:

The section titles have been corrected.

47. Page C-6 (repeated in December 3, 2010 Peer Review Memo, p. 9)

Comment:

My comment 2 in the August 5, 2009 comment set suggested benchmarking MELCOR against the TMI-2 accident. The comment response said validation against TMI-2 would be of limited benefit considering the accident sequences of interest to SOARCA. I think this resolution misses the point. The TMI-2 accident is a very important and useful benchmark on core damage progression and fission product release to the primary system and containment (which in turn determine much of what happens later in time in the accident), and it would be a good idea to benchmark the revised MELCOR model (e.g., hemispherical lower head) against TMI-2. This may not be practical as part of SOARCA, but it would be valuable and should be considered longer term.

Related Comment (December 3, 2010 Memo, p. 7)**:**

Appendix B – Comment 28: The resolution to my comment on the need for TMI-2 benchmarking for MELCOR misses the point. The resolution stated that, "Validation against the TMI-2 event which had a very limited release would also be of limited benefit considering the accident sequences of interest to the SOARCA project." The TMI-2 accident is a very important and useful benchmark on core damage progression and fission product release to the primary system and containment (which in turn determine much of what happens later in time in the accident), and it would be a good idea to benchmark the revised MELCOR model (e.g., hemispherical lower head) against TMI-2. This may not be practical as part of SOARCA, but it would be valuable and should be considered longer term.

Resolution:

The original comment resolution indicated validation against TMI-2 would be of limited benefit considering the accident sequences of interest to SOARCA. The SOARCA team reconsidered your comments and agrees that the TMI-2 accident is indeed a very important and useful benchmark. The MELCOR Code Manual Volume 3 (SAND2001-0929P) contains a collection of benchmark and validation analyses with MELCOR Version 1.8.5. Central values of key MELCOR parameters are derived from the assessments in MELCOR Volume 3. Chapter 13 of this code manual volume is a comparison to TMI-2. (See also the resolution to Dr. Henry's Comments 34 and 38.)

48. Page C-6 (repeated in December 3, 2010 Peer Review Memo, p. 9)

Comment:

The resolution to Comment 49 on the Summary document says that a short paragraph was inserted in the Executive Summary to describe the fraction of emergency phase risk within 10 miles that is attributed to the non-evacuating cohort. I could not find such a paragraph in the Executive Summary.

Resolution:

Consequence results for the nonevacuating cohort will continue to be included in the overall consequence calculations and a short discussion of the nonevacuating cohort will be added to the report. In some of the slowly developing sequences, 100% of the emergency phase risk is from nonevacuees.

49. Page C-6 (repeated in December 3, 2010 Peer Review Memo, p. 9)

Comment:

The resolution to Comment 85 on the Summary document says that the Executive Summary has been enhanced to emphasize that the probability of 50.54(hh) mitigation is assumed to be zero for purposes of unmitigated sequences. I could not find this in the Executive Summary.

Previous Comment 85:
For the same reasons as described in my August 5, 2009 Comment 5, some reasonable probability should be assigned to operator failure to implement the 50.54(hh) mitigative measures. If a factor of 10 is assumed as was done in the August 5, 2009 Comment 5, the unmitigated STSBO sequences (two of them) probabilities would decrease to 1E-8 – 5E-8, and the mitigated STSBO sequences (if they were added to the analysis) would be 1E-7 – 5E-7. (cf. detailed comments by Leaver 10/5/09 for frequency estimates)

Resolution:
The inclusion of both mitigated and unmitigated results is an important feature of the SOARCA results and demonstrates the impact of potential mitigation. While addressing this comment would demonstrate, in a more integrated risk sense, the impact of the 10CFR 50.54(hh) measures, the effort would require a risk and human reliability study, which is beyond the scope of the SOARCA project. A probability/frequency is not assigned to mitigation. Rather a sensitivity calculation is provided to show what the consequences would be without mitigation (the “unmitigated” scenarios), which bounds the spectrum of possible consequences with regard to assigning success probabilities to the 10 CFR 50.54(hh) mitigation actions. The Executive Summary has been enhanced to emphasize that the probability of 10CFR50.54(hh) mitigation is assumed to be zero for the purposes of the SOARCA analysis of the unmitigated cases.

50. Page C-6

Comment:
The Appendix B, page 174 footnote states that inertial deposition is expected to be a significant capture mechanism in the LHSI line, and that other mechanisms “were important”. Is “were important” a typo?

Resolution:
The analysis and text describing the ISLOCA are currently being revised as a result of fact check comments received from the licensee in April 2010. It is expected that the new analysis will address both turbulent deposition and inertial deposition (impaction).

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51. Page 7

Comment:

Best Modeling Practices – Comment 11: The resolution to my comment on the term "physically unreasonable" notes that this phrase "has become a term of art" for remote probability or essentially impossible in an LWR severe accident environment (my words). I still believe that "physically unreasonable" will not connote remote probability or essentially impossible to many that will read the SOARCA document, and I urge the staff to provide clarification in the text where this term is used.

Resolution:

This term is generally associated with events that are deemed so infrequent as to not warrant quantification. It is a term that evolved from historical research into steam explosions, and drywell liner melt through.

The SOARCA team reviewed the use of the term, "physically unreasonable," in the SOARCA reports and the team thinks that adequate clarification is provided to convey the intended meaning where the term is used. For example, note the words emphasized in the following quote from Section 4.8.2, "Early Containment Failure Phenomena," in Appendix A: "The issue of Mark I drywell shell (liner) melt-through at Peach Bottom was assessed by the NUREG-1150 molten core-containment interaction panel. The results of expert panel elicitation are reported in NUREG/CR-4551, Volume 2, Revision 1, Part 2, "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, Experts' Determination of Containment Loads and Molten Core Containment Interaction Issues," issued April 1991. The response was uncertain because there were two schools of thought on this issue. Since the completion of NUREG-1150, the NRC has sponsored analytical and experimental programs to address and resolve this "Mark I liner attack" issue. The results of an assessment of the probability of Mark I containment failure by melt attack of the liner were published in NUREG/CR-5423, "The Probability of Liner Failure in a Mark I Containment," issued in 1989 and NUREG/CR-6025, "The Probability of Mark I Failure by Melt-Attack of the Liner," issued in 1993. It was concluded that, in the presence of water, <u>the probability</u> of early containment failure by melt-attack of the liner <u>is so low as to be considered physically unreasonable</u>." (emphasis added)

BRUCE MROWCA

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52. Page 35

Comment:

Recommendation 1: Provide a summary of the changes that are being incorporated in response to the first supporting objective. Consider rewording this objective to reflect a balanced consideration of risk-significant improvements and challenges.
Associated discussion (p.32): …the stated objectives also appear to be positively biased as indicated by the framing of the first supporting objective. This objective addresses "improvements and updates" as opposed to changes not reflected in earlier assessments. Although this reviewer agrees that there have been many improvements, there are also significant challenges associated with areas such as new fire model methods, increased on-line maintenance or generic issues such as the sump screen issue. A balance discussion should be included in the SOARCA report of the method used to select the changes for incorporation into this project in order to inform the reader as to potential limitations that may not have been addressed.

Resolution:

Improvements in fidelity and modeled phenomena or state of knowledge are not presumed to lead to more favorable predictions of risk. Rather, improvements are assumed to lead to more accurate or realistic predictions. Modeling refinements are neutral with respect to outcome and include effects that can lead to either larger or smaller releases. The SOARCA study was not biased to reflect only effects that would reduce predicted source terms. For example, plant representations that include higher burnup fuel inventories likely lead to larger absolute releases (the second supporting objective mentions this as, "Incorporation of plant changes, such as power uprates and higher core burnup, that are not reflected in earlier assessments;…"; see also the response to Dr. Henry's Comment 31). Good accounting of deposition surfaces could lead to lower absolute releases. Overall, the intent is to render more realistic predictions which the SOARCA team deems a balanced approach.

The SOARCA team attempted to accurately reflect plant conditions. The latest Level 1 information concerning the initiators derived from the plant-specific SPAR models has been used and these are not believed to be biased. Additionally, realistic information such as fuel burnups, power uprates and contemporary higher population densities, all of which have the effect of increasing negative consesequences, are accounted for as well in this intended best-estimate treatment.

53. Page 35

Comment:
Recommendation 2: Provide a better justification for the selected screening criteria. Associated discussion (p. 33): The case for using the selected screening process is not well made. The analysis states that the priority of the work is to bring a “more detailed, best-estimate, and consistent analytical modeling to bear in determining realistic outcomes of severe accident scenarios” and concludes that the benefits could most efficiently be demonstrated by applying these methods to a set of the more important severe accident sequences. However, the stated project objectives are much farther reaching than demonstrating the benefits of realistic analytical methods. The benefits of realistic analysis can be achieved by selecting any relevant set of sequences. For the narrow objective of demonstrating the benefits of realistic methods, this reviewer agrees that approach taken is sufficient. However, the other identified objectives suggest that it is necessary to capture all or a significant portion of the risk. Specifically, a more comprehensive approach would appear to be called for in order to communicate risk and to provide an update of the quantification of offsite consequences contained in NUREG/CR-2239.

Resolution:
Section 2.1, “Approach,” of the Main Report discusses the approach the SOARCA team used, which considered accident scenarios (sequence groups, rather than individual sequences) based on both likelihood and potential consequences. Core-damage sequences from previous NRC and licensee PRAs were identified and binned into core-damage groups. A core-damage group consists of core-damage sequences that have similar timing for important severe accident phenomena and similar containment or engineered safety feature operability. The *groups* (not individual sequences) were screened according to their approximate core-damage frequencies to identify the most significant groups. Since core-damage groups, i.e., scenarios, were considered, many individual lower-order sequences would be captured in the aggregation into groups.

The SOARCA team used selection criteria based on core damage frequency (based on the latest NRC SPAR models) and initiating events involving containment bypass or leading to an early failure of the containment with potential consequences. Scenarios (sequence groups) with a frequency at or above 10^{-6} per reactor-year were screened in, as well as scenarios with a frequency at or above 10^{-7} per reactor year for scenarios that are known to have the potential for higher consequences (e.g., containment bypass scenarios such as steam generator tube rupture and interfacing system loss-of-coolant accident [ISLOCA] initiators)

The selected scenarios for SOARCA were shown to be important in recent and past probabilistic assessments, particularly for the well-studied pilot plants of Peach Bottom and Surry. The selected scenarios are also representative of broad classes of transients. The selection of station blackout (SBO) events in SOARCA ensures that we have covered that broader class of transients involving a loss-of-heat removal and further, by including a short-term blackout we have reasonably bounded that class of accidents (which could include other events such as loss of service water or loss of component cooling water but which develop more slowly). Also, for the PWR, the station blackout also includes, in part, the effect of a small loss of coolant by considering reactor coolant pump seal leakage. In addition, by the selection of station blackout sequences for analysis, we also include the effects of loss-of-containment heat removal (fan coolers) and loss-of-containment spray systems (which are all electrically powered) to remove airborne radionuclides.

The purpose of SOARCA was to do a best estimate consequence analysis. Pursuing a Level 3 PRA which would capture consequences from additional lower order sequences would necessitate introducing conservative assumptions in order to complete analyses and would have served a different purpose.

In addition, the most recent Level-3 PRA-type information we have, while somewhat simplistic and approximate, does indicate that SOARCA is likely to have considered accident scenarios that account for the majority of risk at both the Peach Bottom and Surry plants. The Severe Accident Mitigation Alternatives (SAMA) analysis submitted with the license renewal application (2001) for Peach Bottom indicated that station blackout scenarios accounted for the majority off-site risk (Reference: Peach Bottom, License Renewal Application, Environmental Report, Appendix G, available at:
http://www.nrc.gov/reactors/operating/licensing/renewal/applications/peach-

bottom/peach_bottom-envg.pdf). The SAMA analysis for the Surry license renewal application (2001) indicated that SGTR and ISLOCA accounted for the majority of the off-site population dose risk (Reference: Chapter 5 of NUREG-1437, Supplement 6 (2002),), "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Regarding Surry Power Station, Units 1 and 2 – Final Report," available at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1437/supplement6/2671bpart2.pdf).

Additional analyses and discussion were added to Chapter 7 of Appendix A to include a much more detailed comparison of NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," to SOARCA, and this will be presented at the upcoming planned Peer Review Committee meeting.

54. Page 35

Comment:

Recommendation 3: Provide the risk profile that is being assumed for the assessment of each plant. Although it is understood that this profile is estimated and is developed based on multiple models, it is impossible to judge the degree of risk being capture by this analysis without a clear starting point.

Associated discussion (p. 34): This reviewer believes that the targeted sequences identified in the SOARCA report represent significantly less than the 95% ASME PRA criterion… to obtain the identified screening criteria would require a significantly lower screening value, at least one order of magnitude lower, than that used by the SOARCA Project. The use of the acceptable surrogate goal for the quantitative health objectives contained in the Commission's Safety Goal Policy statement as opposed to the estimated CDF associated with each plant, likely results in significant risk being screened.

Resolution:

SOARCA is not presuming any initial risk profile, nor does it seek to quantify change in plant risk (the resolution to Comment 53 above discusses the scenario selection process). The SOARCA study is not a traditional PRA which is intended to quantify total risk from all sequences, and hence the American Society of Mechanical Engineers PRA criterion was not applied. However, as noted in the resolution to Comment 53 above, the SAMA analyses contained in the 2001 license renewal applications and associated final supplemental environmental impact statements indicate that SBO scenarios account for the majority of offsite risk for Peach Bottom, and ISLOCA and SGTR scenarios account for the majority of the offsite population dose risk at Surry.

55. Page 35

Comment:

Recommendation 4: Ensure that the presentation of accident sequences is consistent between the executive summary and the appendices.

Associated discussion (p. 34): The SOARCA Executive Summary shows that four accident sequences were selected for Surry's consequence analysis with three identified as external event related and one identified from the internal events PRA. The total frequency of these events is $2x10^{-5}$. Appendix B contains some variations to this list including an additional internal sequence associated with a spontaneous steam generator tube rupture (SGTR).

Resolution:

The Executive Summary has been revised, and the discussion of Surry sequences should now be consistent.

56. Page 35 (SOARCA peer review report)

Comment:

Recommendation 5: Provide explicit mapping of the sequences from the set of initial sequences for those that met the screening criteria to those that were considered in the consequence analysis. Ensure that the frequency for each sequence is explicitly identified. Ensure that the reason for elimination of a sequence is clearly stated.

Recommendation 6: Account for all significant sequences.

Associated discussion (p. 34): A review of the internal event sequences contained in the Surry SPAR Model shows that the two internal event sequences selected for the SOARCA Project represent less than 15% of the internal events contribution to core damage and that depending on the approach used to bin the accident sequences several other sequences may have candidates for inclusion in the consequence analysis even if the 10^{-6} criterion was used … Some of these sequences may be considered to have been bounded by the long-term and short-term station blackout (SBO) scenarios, but as currently written, these blackout scenarios appear to be addressing external event challenges and are separate from the internal event-related sequences.

Resolution:
Additional text and references have been added to the reports about the scenario selection process. The scope of the SOARCA project in terms of number of accident sequences was intended to be representative, and was intentionally not organized to be a comprehensive risk study.

The SOARCA team contends that the selected sequences are representative in terms of reflecting the range of potential consequences. We feel that the scenario selection process for SOARCA assures that the more important severe accident scenarios are captured in terms of consequences (especially with the addition of scenarios that would have been screened out based on frequency but included due to historic interest), even though the study is not a traditional PRA. Modeling the consequences of SBO scenarios initiated by external events (e.g., seismic) should be bounding for the consequences from internally-initiated SBO events, since the externally-initiated events assumes the unavailability of additional equipment disabled due to the initiating event.

57. Page 36

Comment:
Recommendation 7: Define the sequence framework that is being used in the SOARCA Project. Ensure that it is consistent with the screening criteria.

Resolution:
As described previously, the SPAR models for the subject plants were used to identify candidate scenarios in the SOARCA framework. The scenario selection process is discussed in response to Comment 53 above.

58. Page 36

Comment:
Recommendation 8: Provide a summary table within each appendix that identifies each sequence meeting the screening criteria, and its treatment within the accident progression and the emergency response sections. Give each sequence a unique identifier and address it in the same order within each section.

Resolution:

Section 2 of the Main Report describes the scenario selection process in detail. Short discussions are also being added to the Appendices. The number of scenarios in the SOARCA project is not extensive; the SOARCA team does not see the need to assign each scenario a unique identifier. The Main Report and Appendices are being reviewed to ensure that scenarios are addressed in a logical and consistent order.

59. Page 36

Comment:

Recommendation 9: Include the identification and/or development of each sequence frequency within Section 3 of each appendix.

Resolution:

The accident scenarios (sequence groups) were developed as an initial step in the project following criteria that was applicable to both plants. The information on the accident scenarios is presented in Section 2, "Accident Scenario Selection," of the Main Report. As explained there, scenario frequencies were based on the latest NRC SPAR models and discussed with each licensee. Further work quantifying frequencies was not part of the SOARCA project.

60. Page 36

Comment:

Recommendation 10: Perform a human reliability assessment for the identified security-related mitigation improvements or identify a conservative screening value so that all sequence frequencies can be calculated.

Resolution:

SOARCA does not include a quantitative human reliability analysis; however, the 10 CFR 50.54(hh) procedures and training were inspected as part of security assessments with site-specific evaluations prepared. Additionally, SOARCA staff performed follow-up site visits in June and August 2010 and January 2011 to explicitly address Reactor Core Isolation Cooling (RCIC) blackstart and run for STSBO and manual operation of TD-AFW and to discuss fact check comments. The site visits included a review of RCIC blackstart and run procedures, additional tabletop exercises to address conservatism in the assumed PWR STSBO timeline, and walkdowns and detailed reviews of procedures (see discussion in Appendix A Section 2.3

and Appendix B Section 1.5). In addition, the recent post-Fukushima inspections at Peach Bottom and Surry provide some additional information with respect to the two plants' abilities to implement SAMGs and B.5.b measures (the inspection reports are available at: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/follow-up-rpts.html). The SOARCA team is currently evaluating the information in these inspection reports for any relevant insights.

A probability/frequency is not assigned to mitigation. Rather, a sensitivity calculation is provided to show what the consequences would be without mitigation, which bounds the spectrum of possible consequences with regard to assigning success probabilities to the 10 CFR 50.54(hh) mitigation actions. Further, this "unmitigated" scenario specific risk quantification assumes the probability of 10CFR50.54(hh) mitigation is zero, which is conservative and bounding. See also resolution to Comment 61.

61. Page 35

Comment:
Conclusions: It is clear that the insights gained from the integrated phenomenological analysis using self-consistent scenarios are significant and the report demonstrates the benefits of this more realistic treatment. However, the focus of this review was on the process for selecting the scenarios and on applying the security-related recovery actions. These activities appear to have serious limitations. The scope of changes considered by SOARCA was not clearly stated, the starting risk profiles of the selected plants was not provided, the appropriateness of the sequence screening criteria was not well defended, the calculation of the sequence frequencies was incomplete and a state-of-the-art human reliability analysis of the security-related actions was not performed. These weaknesses reduce the confidence that all of the stated objectives were met.

Resolution:
SOARCA is demonstrating new approaches to evaluating consequences of severe accidents and is not intended to be a full scope PRA. The intent of SOARCA was to provide a more realistic consequence assessment of traditionally studied accident sequences that are important; it was not the intent to quantify any change in baseline risk of the plants. The SPAR process and the applied screening criteria produced a spectrum of accidents that are similar to (or consistent with) the severe accidents traditionally used to inform source term categories in PRAs. One important feature of the SOARCA approach is that the sequences are examined in

significant detail for realism and not subjected to simplifications inherent in traditional PRA. Realism was emphasized over conservatism.

To the extent possible, human actions were considered, especially with regard to mitigation actions, and when it was considered that insufficient time was available to implement mitigations, they were not credited. The project did discuss the state of human factors sciences and mitigations for which there are no currently well developed procedures. It was a pragmatic decision to explore mitigation actions nevertheless, considering that they would be attempted. A probability/frequency is not assigned to mitigation, rather, a sensitivity calculation is provided to show what the consequences would be without mitigation, which bounds the spectrum of possible consequences with regard to assigning success probabilities to the 10 CFR 50.54(hh) mitigation actions. As noted for Comment 60, the quantification assumed the likelihood of 10CFR50.54(hh) mitigation was zero.

While SOARCA does not include a quantitative human reliability analysis, the 10 CFR 50.54(hh) procedures and training were inspected as part of security assessments with site-specific evaluations prepared. Additionally, SOARCA staff performed follow-up site visits in June and August 2010 and January 2011 to explicitly address licensee fact check comments. The site visits included a review of RCIC blackstart and run procedures, additional tabletop exercises to address conservatism in the assumed PWR STSBO timeline, and walkdowns and detailed reviews of procedures (see discussion in Appendix A Section 2.3 and Appendix B Section 1.5). In addition, the recent post-Fukushima inspections at Peach Bottom and Surry provide some additional information with respect to the two plants' abilities to implement SAMGs and B.5.b measures (the inspection reports are available at: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/follow-up-rpts.html). The SOARCA team is currently evaluating the information in these inspection reports for any relevant insights.

Future PRA could be evolved to include such features and future reliability studies performed to establish success frequencies for risk quantification purposes.

KEVIN O'KULA

Comments dated May 17, 2010

62. Page 3

Comment:

While satisfactory MELCOR- WinMACCS integration was apparently achieved, much of this work was not documented to the appropriate level of detail that would be desirable in a study of this magnitude. More has been said about the integration in Section 6. However, the chronological treatment applied in the SOARCA analysis was notably consistent from scenario selection through offsite consequence evaluation for each of the eight baseline, accident sequences discussed in the NUREG report.

Resolution:

A separate document to describe the MACCS2 methods and choices used in SOARCA is planned (MACCS2 best practices for SOARCA) and should resolve this comment. This document will be published at a later date (a draft is expected to be available in late 2012).

63. Page 3

Comment:

In general, the overall technical results are well substantiated and explained in sufficient detail so as to support key findings and study insights. While good use is made of the NUREG/CR-2239 (Ref. 3) SST1 source term with respect to the composition, timing, and magnitude of the release relative to SOARCA source terms, the opportunity should be seized to connect with Peach Bottom and Surry results from NUREG-1150 where practicable. The SOARCA study is an opportunity to build on the discussion from the landmark severe accident risk study for Surry and Peach Bottom to show how improvements in methods, training, modeling, plant improvements, have substantially reduced severe accident risks. This comparison would be highly informative for those accident sequences, e.g., station blackout or LOCA that were analyzed throughout the 1970s - 1980s and have been revisited during the SOARCA study.

Resolution:

As the comment indicates, we currently give no comparisons with NUREG-1150, at least for the consequence results. While the SOARCA team sees the benefit of this recommendation, comparisons would be difficult. Source terms in NUREG-1150, “Severe Accident Risks: An

Assessment for Five U.S. Nuclear Power Plants," were grouped into source term categories in a way that makes it impossible to trace them back to the initiating event. For this reason, direct comparisons with NUREG-1150 have not been attempted.

As part of Appendix A (Section 7.3.6, "Comparison with Sandia Siting Study"), additional analyses and discussion were added to the report to directly compare the results from NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," to SOARCA. Specifically, since the 1982 Siting Study does not provide latent cancer results at distances that are meaningful and comparable to those provided in the SOARCA study or to the NRC safety goal, an effort was made to reproduce the 1982 Siting Study results for Peach Bottom using the SST1 source term in order to produce results that are directly comparable to the SOARCA results. A comparable write-up will be added to Appendix B for Surry.

The purpose of SOARCA is to do a best-estimate analysis on analyzed scenarios. The overall objective is to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. One supporting objective is to incorporate the significant plant improvements and updates not reflected in earlier assessments, and another supporting objective is to evaluate the potential benefits of recent security-related mitigation improvements in preventing core damage and reducing an offsite release should one occur. SOARCA is not intended, however, to quantify the risk reduction from improvements.

64. Page 5 (Repeated in December 3, 2010 Peer Review Memo, p.10)

Comment:

MELCOR-to-MACCS2 transition - The documentation in the four-volume NUREG report, and especially in Volume I, Summary report, is sparse with respect to the MELCOR to MACCS2 transition. It is difficult to judge how best-estimate aspects of the source term description are based the description provided for deposition velocity and surface roughness length. In the Summary report, the discussion (pages 60-61) is ambiguous regarding the approach to assigning deposition velocity to aerosol particle sizes. Specifically, it is unclear if binning associated with particle size and deposition velocity uses expert elicitation and MELMACCS methodology or if one approach was primary and the other supplementary. The pedigree of the MELMACCS technical report (Ref. 47) appears to be at an internal laboratory report. It is recommended that the report be formally released with adequate technical review.

Resolution:
Additional text has been added to Section 5.4, “Source Term Evaluation,” of the Main Report which discusses the MELMACCS treatment of deposition velocity and aerosol particle sizes.

Providing an explanation for the connection between MELCOR and WinMACCS would be beneficial for MACCS2 modelers who would like to update the way they import source term data from the methods used at the time of NUREG-1150, which is still common practice among MACCS2 modelers. A general description of the connection between MELCOR and WinMACCS will be added to the MACCS2 best practices for SOARCA document, which will be published at a later date (a draft is expected to be available in late 2012).

65. Page 5

Comment:
Surface roughness length (z_o) - The documentation in the Volume I Summary is very brief and omits much technical justification for selection of key parameters. One of the areas that remains too limited is the basis for the surface roughness length (z_o). Ten centimeters surface roughness length would seem to be overly conservative for Peach Bottom given the fifty-mile environment surrounding the plant. The same value may be appropriate for Surry but a more complete discussion is recommended. Is the 10-cm value used for both plants an indication that the environments around the plants were considered characteristic of tall grass (refer to Table 2.3 from NUREG-4691 Vol. 2 and shown in Figure 1 below)?

Resolution:
The 10 centimeter (cm) value for surface roughness has been used as a typical value for the US and site-specific values have generally not been considered in previous analyses, such as NUREG-1150. To address the choice of surface roughness, a brief discussion of the selection of the 10-cm roughness length has been added to Sections 5.4, “Source Term Evaluation,” and 5.5, “Site-Specific Parameters,” of the main document. Also, a sensitivity study varying surface roughness is currently being performed, and we will include the results in the SOARCA best-estimate report Appendices. The sections we are adding to discuss these sensitivity analyses will provide a more detailed discussion of the selection of site-specific surface roughness for Peach Bottom and Surry. We also plan on addressing this during the uncertainty analysis for the project. See additional discussion in resolution for Comment 66.

66. Page 6

Comment:
Deposition velocity selection - The deposition velocities associated with various aerosol sizes is given in Table 13 of the Summary (page 61). However, it is not clear what radionuclides groups are associated with one or more of the bins shown in Table 13 and how the median diameter bins would be distributed for a given radionuclide group. Surely a state-of-the-art input distribution of deposition velocities would offer a different distribution based on physicochemical characteristics of one group vs. those of another.
For example, the halogen isotopes (primarily radioiodine) will deposit differently from cesium species (e.g., CsI and CsMo04). It is suggested that while realistic input is reflected in the SOARCA study and used in the current analyses, justification should be provided to support its use in place of the sample input published in the 1998 MACCS2 User's Guide (Ref. 5) for the allocation fractions for nine fission product groups (Figure 2).

Related Comment (December 3 Memo, p. 11):
It is suggested that while realistic input is reflected in the SOARCA study and used in the current analyses, justification should be provided. A draft copy of Ref. 48 (SOARCA Summary document) seems to indicate that deposition velocity was based on an expert-elicitation approach (page 43), with prairie, forest, and urban surface roughness length used as a parameter by the experts. The overall process that was applied is not apparent and it would be greatly benefit the intended NUREG documentation if additional detail could be provided.

Furthermore, I am apparently in the minority with the approach used but believe it paints a picture that deposition would occur in the same way regardless of the site of interest. Surry and Peach Bottom may be sufficiently alike to justify this approach for SOARCA, but in general I don't believe it is correct for smaller size aerosols that interact with the surface features, and as such, are not controlled by settling velocity alone.

Resolution:
Section 5.4, "Source Term Evaluation," of the Main Report describes how typical values for surface roughness and mean wind speed, 0.1 m and 2.2 m/s, respectively, are used as inputs to MELMACCS in order to calculate deposition velocity. Mean wind speeds were determined from the specific weather files used in the consequence analyses. Table 3 of the Main Report displays the deposition velocities used in SOARCA analyses for both Peach Bottom and Surry.

In the MELCOR treatment of aerosols the size distribution is discretized into size intervals or sections based on the “sectional treatment” of the MAEROS model. Within each size section, all particles are identical and each particle is considered to be composed of proportional fractions of the chemical groups present as aerosols in a particular volume. Micro-photographs show particles are agglomerations of many small particles. Even the smallest of these, which are sub-micron are comprised of many elemental forms. Since they are formed as condensation nucleii from co-mixed vapors, there really isn't any way for them to partition into distinct pure particle species. Since most of MELCOR aerosol size bins are populated as fission product vapors are condensing and particle number densities are very large, agglomeration rates are therefore very large. In that case, there is no way to keep them apart and it is a very good approximation.

At the same time, the fractional composition of the chemical classes within each size section varies over the transient and the aerosol size distribution itself also changes over the course of the transient. Because of this, the average (over the time of the accident) particle size distribution is different for each chemical class. Using the transient information in the MELCOR plot file, MELMACCS calculates a particle size distribution for each chemical class using the MELCOR sectional class fraction information, and uses that to define an independent size distribution for each chemical class.

A brief description of the calculation of aerosol size distributions is now included in Section 5.4 of the Main Report. An expanded description is planned to be included in a separate document that describes SOARCA modeling choices and methods for consequence analysis (MACCS2 best practices for SOARCA), which will be published at a later date (a draft is expected to be available in late 2012).

MACCS2 uses the aerosol size distributions by chemical group, plus the deposition velocities in the table, to determine the rate of depletion of aerosols from the plume. Generally, the larger aerosols deposit more quickly and so are depleted more rapidly from the plume. The peak in the aerosol size distribution is usually a few micrometers, which corresponds to a deposition velocity of a few millimeters per second.

The sensitivity study to explore the effect of surface roughness, mentioned in the above response to Comment 65, will also include the effect of surface roughness on deposition velocity. So, the study will include the effects of surface roughness on vertical dispersion and on deposition velocity for both the Peach Bottom and Surry sites. The results of the sensitivity study will be folded into Chapter 7 of Appendices A and B.

67. Page 6 (repeated in December 3 memo, p. 11)

Comment:

Discussion of the non-site specific and site-specific parameters used: A key outcome of the SOARCA project, discussed several times with the Peer Review Panel during our meetings, has been the smaller and delayed source term compared to the Sandia Siting Study. These important results are illustrated in Table 21 (Appendix A – Peach Bottom Integrated Analysis) and Table 32 (Appendix B - Surry Integrated Analysis). Especially in the case of the Surry conditional risks with distance (Figures 172, 174, 176, 178, and 180), the results are dominated by the long-term (CHRONC) phase of the analysis. This aspect of the model is driven by user input of dose criteria (habitability dose limits), site-specific data, and some non-site specific information. It would improve the understanding of the SOARCA study to provide the tables of information to augment the discussion of sections 5.5 and 5.6. The tables would mostly address the type of inputs that control the EARLY and CHRONC modules, specifically in deciding on condemnation and inhabitation return and be as follows (with suggested location by volume):

Table 1. Peach Bottom Offsite Consequence Analysis Site-Specific Parameters (Appendix A)
Table 2. Surry Offsite Consequence Analysis Site-Specific Parameters (Appendix B)
Table 3. Site-Independent Parameters (Summary – Section 5.6).

While some of these data (e.g. shielding factors) can be found elsewhere in the four-volume set of SOARCA documents, it would assist the reader to see the key inputs collected in one or several tables. This information would be useful in determining what sensitivity studies are important, what the focus should be on plant features, severe accident mitigation procedures, etc., and to what extent public policy has a role.

Resolution:

The recommended tables for the input parameters will be added in the SOARCA reports.

68. Page 7

Comment:

Boundary weather – During one of the review meetings, it was indicated whether a boundary weather condition is imposed, with forced deposition conditions, and if so, the type of weather specified and at what region in the grid. I don't think this is established in the February final draft for review document. This aspect of the offsite consequence analysis should be described for comparison with past work (e.g., NUREG-1150).

Resolution:
A brief discussion on boundary weather will be added in Section 5.5, "Site-Specific Parameters," of the Main Report. In all of the SOARCA calculations, boundary weather is applied beyond the 50-mile region for which risk is characterized. Thus, the conditions chosen for boundary weather have no influence on the reported results.

69. Page 7 (repeated in December 3 memo, p. 11)

Comment:
Centralized discussion of MACCS2 improvements- There are many improvements noted in the MACCS2 model with 64 directional sectors and more realistic evacuation modeling among others. It would be informative to have a short section in Appendix A and Appendix B to summarize the prominent features and expanded capabilities by module, i.e., ATMOS, EARLY, and CHRONC. This version of the code has expanded capabilities for performing uncertainty analysis but little is covered in the documentation or was commented upon in the three presentations.

Resolution:
Main Report Section 5.0, "Offsite Consequence Analyses," has a short discussion of the modeling improvements and related prominent features and expanded capabilities used in SOARCA, and additional references to Main Report Section 5.0 will be added to the Appendices.

70. Page 7

Comment:
Reporting of additional consequence measures – In addition to the conditional and absolute early/latent health effect risks reported in the SOARCA study, other metrics would be advised. The uncertainty quantification and sensitivity analysis presentation in March 2010 indicated that land contamination was being considered, and would be very useful. It is advised to clarify

whether forced deposition is used, and if so, the inner and outer radii the feature is employed. To compare to earlier studies, the metric of population dose over the fifty-mile region would be a useful consequence measure to complement land contamination. This is appropriate because deposition with distance would trend inversely with inhalation doses.

Related Comment (in December 3 memo, p. 10)**:**
Reporting of additional consequence measures – In addition to the conditional and absolute early/latent health effect risks reported in drafts of the SOARCA study, other metrics are advised. To be able to compare to earlier studies, the metric of population dose over the fifty-mile region would be a useful consequence measure to complement land contamination. The uncertainty quantification and sensitivity analysis presentation in March 2010 indicated that land contamination was being considered, but nothing to my knowledge was covered on this topic in the October meeting. The lower deposition velocities used in SOARCA compared to past studies, have implications for both populations dose and the land contamination consequences.

The reduced and delayed source terms compared to SST1 from the Sandia Siting Study, in the cases considered from SOARCA, provide Level 2 insights. Examples of these comparisons are shown in Tables 21 and 32 in the plant-specific reports for Peach Bottom and Surry, respectively. In addition, the LCF risk comparison of SST1 to the Peach Bottom and Surry, and other SST1 results (Peach Bottom: Figures 89 and 90; Tables 30-32) (Surry: Figures 181 and 182; Tables 46-48) accidents are useful for bringing together improvements to the health effects models, as well as smaller source terms and better modeling of countermeasures. Adding land contamination and populations dose metrics out to the 50-mile radius would produce a more immediate measure of improvements to our consequence understanding and bridge the gap from the Level 2 to the latter parts of the Level 3 insights.

Resolution:
There is no forced deposition used within the modeled 50-mile region. Historically, deposition velocities were overestimated. Current values are recommended based on the past separate MACCS peer review.

Additionally, in Appendix A, a comparison to SST1 was added to the report (as noted in the resolution to comment 63 above).

The SOARCA team recognizes that other metrics may have value, however, the scope of SOARCA has been focused on those metrics which we believe to be most important (e.g., public health effects expressed as individual risk, which is the metric that the NRC uses for its quantitative health objectives [QHOs]).

December 3, 2010 Peer Review Memorandum

71. Page 11

Comment:
Uncertainty analysis study – There are many major improvements that the SOARCA study has discussed during the Peer Review meetings and have been documented in the NUREG drafts. However, I fear that discussion of these improvements will be met by a high level of technical questioning that can only be satisfied with a comprehensive uncertainty analysis. This is a major drawback in the current study's body of information and ought to be completed before the technical reports are finalized.

Resolution:
The uncertainty analysis is underway and draft results are expected to be available in mid 2012. The numerous sensitivity analyses that have been completed, of which many results will be included in the SOARCA Main Report and Appendices, should provide confidence in the best estimate results.

72. Page 12

Comment:
Presentation and labeling in the SOARCA Documents - To improve the likelihood that the public will interpret the SOARCA study as intended, several recommendations follow:
(1) Aim for uniformity and consistency in the labeling of conditional and absolute risk figures and tables. I suggest conditional risk per event (LCF/event) and for absolute risk (LCF/reactor-year). Currently, both types of risk are labeled the same in figures and tables.
This situation may lead to incorrect interpretations by the reader. (2) Label LCF bar charts with Acute phase and Long-term phase rather than EARLY and CHRONC. The analysis is using MACCS2 as a tool to evaluate the relative importance of the short-term and long-term phases, and it should be made transparent that this is the case. Use of MACCS2 terminology in the results gives the appearance that the results are more characteristic of the manner in which the code was run, and not reflective of the post-release phases. (3) Select two of the four health

effect (dose truncation) models rather than present results from all four models. The LNT model appears to be bounding in all cases. It is recommended that this model be retained along with the one that tends to predict the most non-conservative health effect risks of the three alternative dose truncation models, i.e.,

- Health Physics Society recommendation (5 rem/year and 10 rem lifetime)
- ICRP Report 104 (10 mrem/year)
- U.S. Average Background (620 mrem/year).

Labels to figures and tables should reflect the dose truncation models with a short-hand notation of “LNT”, “HPS”, “ICRP 104”, and “U.S. Bkg. Ave.”.

Resolution:

The SOARCA team plans to modify the presentation of results as recommended, with the exception that we will retain 3 of the dose models. The 10 mrem dose truncation results are similar to the LNT and are also always slightly less than the LNT results which is why they are not included. A note explaining this will be added to the report. The model that gives the lowest latent cancer fatality risk varies (Health Physics Society recommendation or U.S. Average Background).

JOHN STEVENSON

April 30, 2010 SOARCA Peer Review Draft Report

73. Page 38

Comment:

A basic concern in his evaluation is the potential for liquifaction of soil or other foundation failure associated with seismic induced cyclic motion resulting in large vertical differential displacement of the containment or adjacent structures resulting in rupture or significant leakage of one or more of containment penetrations. A secondary consideration, and with much less probability, is the development of a fissure in the foundation media under the containment or adjacent structure propagating to the surface below the containment or other power plant structures resulting in their foundation failure. The potential for liquification induced failure is limited to saturated cohesionless soils while potential fissure failures are not so limited. In addition to containment penetration failure, fissure type failure if credible might cause simultaneous failure of the containment basemat and supports of the reactor coolant system SSC. It is understood that the types of foundation failure just described resulting from earthquakes at the median 10^{-5}/yr or mean 10^{-4}/yr probability of exceedence level have been negated by design measures such as use of engineered backfill hence; were not, nor should they have been, considered in design.

Related Comment (p. 39)**:**

Given that the site would liquefy, it would be necessary to evaluate the effect of such liquification on the leak tightness of the containment.

Related Comment (p. 40; repeated in part in December 3, 2010 memo)**:**

There may also be other NPP sites where liquification and associated differential displacement, which could cause containment penetration failure which could provide a significant containment leak path, cannot be ruled out at the 10^{-6} or 10^{-7}/yr seismic probability of exceedence levels.

As a result of the potential for liquefaction at the Surry site, it is my recommendation that a follow up on the SOARCA study be conducted which considers seismic induced soil liquefaction, consolidation and possible foundation failure which could lead to early containment be conducted. The primary concern associated with liquefaction or consolidation is that differential settlements of the containment or adjacent buildings may exceed the capacity of even a single penetration to resist significant leakage of the typically more than 100 such penetrations in the containment which could lead to early containment bypass.

Resolution:
NRC is considering future research on soil liquefaction and its effects on nuclear plant structures. The potential effect of increased leakage during the technical specification (TS) leakage phase was considered and it was judged that the effects of the later containment failure overwhelm the effects of any increased leakage during the earlier TS leakage phase. A sensitivity study on increased leakage is included in Appendix A.

74. Page 40

Comment:
A second concern raised by Dr. Stevenson was the potential for hydrogen detonation resulting from DBA hydrogen generation rather than a hydrogen deflagration. Hydrogen detonation studies have indicated a dynamic pressure buildup of 2 to 3 times the containment static design pressure which is at or very near the static failure pressure for the containment. Given the dynamic nature of the detonation this might result in a dynamic load factor depending on duration of the pressure load relative to the period of the containment structure which would amplify the peak pressure loads.

Related Comment (December 3, 2010 Memo, p. 13):
I got the impression at the meeting that a containment hydrogen detonation as compared to a deflagration as the result of a LOCA that could lead to early containment failure was not an impossibility; hence, would be an issue at the very low probabilities of exceedence associated with SOARCA.

Resolution:
Studies were conducted on hydrogen detonation and deflagration and are included in the report (e.g., Section 5.2.3, “Uncertainties in the Hydrogen Combustion in the Mitigated STSBO” in Appendix B). According to MELCOR analysis, hydrogen detonation could produce substantial pressure loads, however the sprays are effective at settling airborne aerosols before detonable quantities could be formed that could fail the containment. The resulting fission product release would consist of only noble gases and would not be expected to substantially increase the

offsite health consequences relative to the base case calculation. Accumulations of hydrogen (>15%) have the potential to produce detonations that, as Dr. Stevenson points out, produce dynamic loads which can exceed the capacity of the containment structure. The consequences of such events are dependent on the quantity of airborne fission products at the time. The Surry sensitivity studies suggested that airborne radioactivity at the time of detonations were in fact low (see Appendix B Section 5.2.3). The SOARCA team will discuss this at the next planned Peer Review Committee meeting.

KAREN VIEROW

April 30, 2010 SOARCA Peer Review Draft Report

75. Page 72

Comment (repeated in December 3, 2010 Memo):

The primary objective of the SOARCA project is stated in several locations of the SOARCA document and in presentations to the Peer Reviewer Committee to be a "best estimate evaluation of the likely consequences of important severe accident events …" The first such claim appears in the Abstract of the Summary Report. Other locations such as the Abstract of Appendix B state that "This study has focused on providing a realistic evaluation of accident progression, source term and offsite consequences…"

It is suggested that the current evaluations are not entirely best-estimate and that care be taken in the SOARCA documents to qualify this claim. A claim to *more best-estimate*, or *more realistic,* results than produced by earlier analyses is appropriate.

While the SOARCA team has done a commendable job of enabling more realistic evaluations of severe accident consequences, several conservatisms have, in fact, been retained. Many of these conservatisms are, in the judgment of this peer reviewer, reasonable and should be discussed collectively in a visible location within the SOARCA document.

One example of conservatism is the assumption for Surry that 8 hours would be required to transport a portable diesel-driven pump and connect it to plant piping following a large seismic event (Appendix B, Section 3.1.3 Mitigative Actions). The licensee staff estimates that 2 hours would be required. A first reading may leave one with the impression that excessive conservatism has been invoked. Upon study of the event timing for mitigated events, one sees that the event sequence does not extend to the containment until 7 hours 16 minutes for the mitigated short-term blackout or 7 hours 30 minutes for the mitigated short-term station blackout with thermally-induced steam generator tube rupture. Containment spray is initiated at 8 hours for these two scenarios and sprays are not needed for the other Surry scenarios. Earlier spray activation should have some effect upon the severe accident progression, with respect to containment pressurization and retention of fission products. Discussion of the conservatism would be useful.

Resolution:

The SOARCA team understands the concern related to the terms “best-estimate” and “more realistic” results. The SOARCA team, however, did strive to seek the “best-estimate” results. See comment resolution 76 for additional discussion.

76. Page 73

Comment (repeated in December 3, 2010 Memo)**:**

Because a “best estimate evaluation” is a stated primary goal of the SOARCA project, this reviewer suggests that a compendium of conservatisms be included within the SOARCA documentation, perhaps as an appendix or within a discussion section on the extent to which SOARCA objectives have been met. Within this appendix or discussion, the argument should be made that inclusion of some conservatism is warranted. Two reasons for justification come to mind. Firstly, conservatism is one method for treating uncertainties. Secondly, if a nonconservative approach were to be taken, the SOARCA project could be interpreted by the public as being overly optimistic about nuclear safety and thereby lose credibility.

Another suggestion is to perform a calculation in which the conservatisms are removed. For example, have the containment sprays operable from two hours and observe the differences in results. This approach is analogous to performing the consequence analysis using actual weather condition from a typical day, instead of specifying conservative or time-averaged conditions.

In summary, care should be taken in public documents and presentations to qualify the degree to which the analysis methods and results can be regarded “best-estimate” or “realistic”. The qualified claim of *more realistic* evaluations seems appropriate

Resolution:

A guiding principal of the SOARCA analysis has been to avoid conservatisms and make every attempt to provide best estimate results. Nevertheless, there are conservatisms that are still reflected in the SOARCA results. For example, the assumption of mid-day population motion during a weekday to present the most challenging evacuation scenario. At the same time emergency response organizations are assumed to be staffed at nighttime levels.

Conservatisms were included in the analyses where it was judged that the additional work necessary to justify a better estimate was not worthwhile, most notably because it was not expected to materially affect the results. While we do not plan to include a compendium of conservatisms in the SOARCA documentation, conservatisms are apparent in the SOARCA reports. For example, Section 6 of appendices A and B discuss the assumption of mid-day population motion during a weekday.

December 3, 2010 Peer Review Memorandum

77. Page 14

Comment:
Referring to the second objective, consideration of power uprates and higher core burnup in the MELCOR analysis is unclear.

Resolution:

The MELCOR decks reflect the current status of the Peach Bottom and Surry plants, and thus reflect the approved power uprates and higher core burnups at those plants (compared to earlier assessments). For the purposes of estimating the available radionuclide inventory, SOARCA analyses assume the accident occurs mid-way through a typical operating cycle. SOARCA analyses used actual core design and performance data for three consecutive contemporary cycles to build the MELCOR and ORIGEN input data. The approach is discussed in the main report Section 4.3.4, “Radionuclide Inventory,” in Appendix A Section 4.7, “Radionuclide Inventories and Decay Heat,” and Appendix B Section 4.5.2, “Initial Radionuclide Inventory.”

78. Page 14

Comment:
Referring to the fourth objective and communication with the general public, a document written in layman’s terms is needed

The SOARCA team has a separate explicit task from the Commission to ensure that SOARCA is communicated effectively to the public. A plain-language brochure will be developed for release at the same time as the SOARCA reports.

JACQUELYN YANCH

Comment report received May/June 2010

79. Page 1

Comment:
We know very little about the health impact of low dose and, more particularly, of low dose-rate radiation; we should make every effort to redress this lack of understanding so that the public can be appropriately guided as they deal with the aftermath of a severe reactor accident.

Related Comment (p. 16):
As mentioned in the SOARCA documentation, little guidance as to how to estimate the likely health impact of low dose, low-dose-rate radiation exposure is provided by the national and international committees who examine available data, and the approach we borrow from radiation protection risk estimates is *not appropriate* for use in dealing with long-term exposures due to radionuclides in the environment. Therefore, while it is not the role of the NRC to dictate how the RHDLs should be set, the NRC and the industry should take a stronger position on determining the true nature of radiation-related health effects at the dose-rates anticipated following a severe nuclear power plant accident. It should be a priority.

Resolution:
The SOARCA team agrees that the state-of-the-art in low dose assessment can be improved; however the team feels that the SOARCA study is consistent with the current state-of-the-art available, and represents an improvement over earlier estimates.

80. Page 2

Comment:
What is the health impact of the return-home dose-rates? We don't know yet. None of the data we use in estimating radiation-induced health effects were obtained at the doses and dose rates similar to those encountered upon returning home. Therefore we have essentially *no* understanding of the potential health consequences of these radiation conditions. More importantly, we have no understanding of the health impact of the radiation dose-rates that were avoided by staying away from home for so long.

Resolution:

In the SOARCA study, the SOARCA team addressed the uncertainty of health effects induced by low doses and dose rates by evaluating a set of dose-truncation values. We believe that at least to some small degree, we have thereby shed some light on this issue and the extent to which it affects risk predictions. In addition, we plan to explore the effect of the return criteria in the uncertainty analysis for the project.

81. Page 16

Comment:

Some countries operate a registry for tracking annual occupational dose for all radiation workers [35]. The existence of such a registry makes it feasible, at some time in the future, to examine health effects as a function of doses received. In most case, however, occupational doses are much *smaller* than individual medical doses [14,29] and therefore any health impact of occupational radiation may never be discernable from the potential effects of the larger medical doses. It makes more sense, therefore, to record our medical doses and to store these in a database. This we do not do.

A couple of decades ago the doses received from diagnostic radiology were relatively small and experienced by relatively few individuals. Today, however, radiological exams are used for addressing a much broader range of medical questions and are performed on a much bigger fraction of the population. More important is the fact that we've begun to make routine use of the more dose-intensive procedures of x-ray computed tomography (CT) and interventional fluoroscopy [17]. The result is that the average US resident receives as much radiation dose from diagnostic radiology procedures as from all natural background radiation sources, combined. Thus, on a routine basis, and for a variety of reasons, we deliberately and carefully irradiate most members of the U.S. population, exposing them to a wide range of doses depending on the reason for the exam, the part of the body being imaged, and the patient's body thickness. It makes sense to maintain a registry of radiation doses for everyone irradiated. This registry would not be a "de-identified" patient radiation dose data-base (as proposed recently by the FDA as a starting point for establishing consistent exam parameters across medical institutions [36]), but a registry that allows tracking of an individual's dose over time and, ultimately, for correlation of dose with disease or health status many years later.

Resolution:

The existence of a database or multiple databases that track occupational and medical doses would further the knowledge on health effects as a function of dose received over protracted periods. The SOARCA team agrees that the state-of-the-art in low dose assessment can be improved; however the team feels that the SOARCA study is consistent with the current state-of-the-art available, and represents an improvement over earlier estimates.

82. Page 18

Comment:

For understanding the impact of chronic, low dose-rate radiation we can examine the many regions of the world whose inhabitants are already living with the dose rates represented by the RHDLs (and higher), and have been for many generations. Residents of high background radiation areas (HBRAs) do not appear to suffer adverse effects from these dose-rates (and in some cases appear to be healthier and living longer than those living in nearby control areas with lower radiation levels [15,37,38]). Such comparisons, however, often suffer from small sample size, incomplete dosimetry, and a lack of uniformity between studies that prohibit combining of the data. Many comparison studies are ecologic in design in which dosimetry data are aggregated over the entire populations; this type of study is subject to several bias and correlation problems which do not arise in cohort or case/control studies where information for each individual subject (dosimetry, confounding factors, etc.) is available [39]. UNSCEAR has suggested that only cohort or case/control studies are suitable for quantification of radiation risk [40].

Resolution:

Dr. Yanch in her comments has identified a number of known limitations in the current dose response models used to predict radiogenic human health risks. Many of these issues arise from the extrapolation of health risks of populations exposed to high-dose, dose-rate radiation—e.g., Japanese atomic bomb survivors— to populations exposed to low-dose and low dose-rate radiation. Future consequence analyses could benefit from research along the lines suggested by Dr. Yanch.

APPENDIX C

SOARCA PUBLIC COMMENTS SUMMARY

The State-of-the-Art Reactor Consequence Analyses (SOARCA) Report (NUREG-1935) was released as a draft for public comments from January 31, 2012, through February 29, 2012. Comments related to the SOARCA project covered a wide range of topics. This appendix provides a summary of the different questions and comments received related to SOARCA, along with NRC responses. Some comments are paraphrased directly from the submitter, while others have been combined and condensed to be more concise. All comments received are located in the NRC's Agencywide Documents Access and Management System (ADAMS) at Accession No. ML12122A946. The comments are related to the following general areas of the SOARCA project:

Project Scope C-1
Scenario Selection..C-3
MELCOR and Accident Progression Analysis C-5
Emergency Response Analysis C-9
MACCS2 and Offsite Consequence Analysis C-12
Public Comment Period and Timing of NUREG-1935 Release..........................C-19

Project Scope

1) Comment:

"The report should address the effects of the accident on the entire site, including additional reactors and spent fuel, not just the one reactor."

NRC Response:

The objective of the SOARCA project was to develop best estimates of the offsite radiological health consequences for potential severe reactor accidents for two plants, Peach Bottom and Surry. Unlike a traditional probabilistic risk assessment (PRA), SOARCA elected to focus its resources on the more important severe reactor accidents, considering both likelihood and potential consequences so that these could be modeled in much greater detail than was done in past studies. This report has noted that its calculated latent cancer fatality risks are for the specific reactor accident scenario only and are not intended to characterize the overall risk to the public posed by operation of the nuclear power plant.

The NRC has established a Site Level 3 PRA project that will address accident effects on the entire site, including additional reactors and spent fuel. Preliminary information on the scope of the Site Level 3 PRA project is available in SECY-11-0089 and SRM-SECY-11-0089.

2) Comment:

"The report avoids characterizing human performance."

NRC Response:

An important objective of the SOARCA project was to assess the impact of severe accident mitigative features and reactor operator actions in mitigating an accident. Rather than conducting a formal human reliability analysis, this was done by evaluating in detail the operator actions and equipment that may be available (including 10 CFR 50.54(hh) equipment) and running accident progression calculations for each scenario twice, first assuming operators are fully successful in carrying out the mitigating actions ("mitigated" case) and then assuming that the plant fails to implement 10 CFR 50.54(hh) measures ("unmitigated" case). By comparing the "mitigated" and "unmitigated" cases for a given scenario, the potential benefits of the post-9/11, 10 CFR 50.54(hh) mitigation measures are evident.

The NRC has established a Site Level 3 PRA project that will include a formal human reliability analysis. Preliminary information on the scope of the Site Level 3 PRA project is available in SECY-11-0089 and SRM-SECY-11-0089.

3) Comment:

"A reactor while in refueling mode only has secondary containment to rely on to protect the public. Why is an accident occurring during this condition not included in the report?"

NRC Response:

Low-power and shutdown accidents are potentially significant because the plant configuration is altered—the containment may be open and the reactor safety systems may be realigned. However, offsetting mitigating attributes include a potentially much smaller decay heat level and low pressure that allows for easier cooling of the reactor fuel. SOARCA has focused on full-power accidents that historically have received the most attention since they have the highest possible decay heat levels and high pressures that limit reactor fuel cooling. Also, one of the objectives was to provide an updated quantification of consequences from past studies, such as NUREG/CR-2239 ("Technical Guidance for Siting Criteria Development," dated December 1982), which were confined to full-power reactor events.

The NRC has established a Site Level 3 PRA project that will include all modes of operation, not just full power. Preliminary information on the scope of the Site Level 3 PRA project is available in SECY-11-0089 and SRM-SECY-11-0089.

4) Comment:

"SOARCA did not include acts of malice for either Peach Bottom or Surry."

NRC Response:

The NRC carried out SOARCA to examine accidents involving mechanical failures and natural events. The NRC's security-related studies conducted after September 11, 2001, however, led the agency to conclude the public would remain safe after a malicious act against a U.S. nuclear power plant. The security-related studies also showed that previous risk studies used conservative radionuclide source terms. The security-related study results suggested improved modeling that accounts for plant improvements would confirm that radionuclide releases and early fatalities were substantially smaller than earlier studies suggested.

5) Comment:

"Uncertainty must be respected by making certain that appropriate and up-to-date methods and assumptions are used in the analysis. SOARCA failed to do so."

NRC Response:

The SOARCA project was a major effort to perform best-estimate calculations of accident progression and radiological health consequences for important severe reactor accidents at Peach Bottom and Surry. The SOARCA team took great effort to ensure that the analyses were based on current plant conditions, equipment, procedures, and current emergency preparedness plans, and used the state-of-the-art in computer modeling. Since this is a very new type of study, SOARCA also includes an uncertainty analysis of one of the scenarios to better understand how the uncertain input parameters affect the accident progression and health consequences. This uncertainty analysis will be completed, documented in a NUREG/CR series report, and made publicly available later in 2012.

Scenario Selection

6) Comment:

"SOARCA did not include extreme natural events which could lead to an early release of radioactive materials."

NRC Response:

The SOARCA study analyzed station blackouts assumed to be initiated by low probability seismic events. The long-term station blackout (LTSBO) is initiated by an earthquake of 0.3-0.5 g peak ground acceleration (PGA). The short-term station blackout (STSBO) is initiated by an earthquake of 0.5–1.0 g PGA. Despite such large seismic initiators, the damage does not cause immediate containment failure and core damage. SOARCA excluded more extreme seismic events that conceivably could cause immediate containment failure followed by core damage. Seismic fragility quantification for these extreme and rare seismic events, in particular quantification of the size of a hole or amount of leakage, is currently subject to considerable uncertainty. More research is needed before undertaking a realistic, best-estimate analysis of such rare events. The probability of such an extreme earthquake affecting Peach Bottom or Surry is sufficiently low that SOARCA elected to focus its resources on the more likely (though still remote) earthquakes in the 0.3–0.5 and 0.5–1.0 g PGA ranges. The external events analyzed in SOARCA are discussed in greater detail in Sections 2.4.2 and 2.5.2 of NUREG-1935.

7) Comment:

"The probability of a severe accident used in SOARCA was far too low; it ignored the real-world lessons from Fukushima."

NRC Response:

SOARCA identified scenarios using the frequencies of severe accidents contained in the best available information sources for U.S. commercial nuclear power plants. These sources included the NRC's plant-specific Standardized Plant Analysis Risk models, licensee PRA, Individual Plant Examination submittals, Individual Plant Examination of External Events submittals, and earlier analyses, such as NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants." Comparisons between the SOARCA plants and the Fukushima Dai-ichi plant should recognize that the seismic hazards at the Peach Bottom and Surry sites are lower than the seismic hazard at the Fukushima Dai-ichi site.

8) Comment:

"SOARCA assumes that tsunamis would not happen at either Peach Bottom or Surry."

NRC Response:

The project team sought to focus its attention and resources on the important severe accident scenarios for Peach Bottom and Surry found in past risk studies, such as NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants." Tsunamis are extremely unlikely for Peach Bottom and Surry. The likelihood of a tsunami reaching Peach Bottom or Surry is sufficiently low that SOARCA elected to focus its resources on the more likely (though still remote) events such as earthquakes.

MELCOR and Accident Progression Analysis

9) Comment:

"MELCOR could have bugs in either the models or the solution methods that result in answers that are incorrect. MELCOR models could be incomplete or not applicable to the problem or have wrong data. Users of MELCOR could be inexperienced and use the code incorrectly. MELCOR does not meet DOE Quality Assurance Standards for Safety Software."

NRC Response:

MELCOR is a fully integrated, engineering-level computer code that the NRC and the international nuclear safety research community has used for nuclear power safety analysis for 21 years. It is a modular code comprising three general types of packages: (1) basic physical phenomena (i.e., hydrodynamics, heat and mass transfer to structures, gas combustion, aerosol and vapor physics), (2) reactor-specific phenomena (i.e., decay heat generation, core degradation, ex-vessel phenomena, sprays, and engineering safety systems), and (3) support functions (thermodynamics, equations of state, other material properties, data-handling utilities, and equation solvers). As a fully integrated code, MELCOR models all major systems of a reactor plant and its important coupled interactions.

MELCOR has been under continuous development by the NRC and Sandia National Laboratories (SNL), where software quality assurance (SQA) is an integral part of the development process. The MELCOR SQA program is adapted from two internationally recognized standards, CMMI and ISO 9001. These standards provide elements of traceability, repeatability, visibility, accountability, roles and responsibilities, and objective evaluation. The MELCOR SQA program focuses on reducing code error, improving documentation of all processes, and continuous integration of procedures into daily work processes. The MELCOR code has been validated against numerous separate effects tests, integral tests such as those conducted at the French Phebus facility, and actual accident studies such as Three Mile Island Unit 2. It has benefited from a global community of users who participate in the annual MELCOR technical review meetings, including MELCOR Code Assessment Program (MCAP) and the European MELCOR User Group (EMUG). Bugs continuously have been reviewed, and SNL has improved the code in response. Sandia has addressed many, if not all, of the important shortcomings identified by the U.S. Department of Energy (DOE) "MELCOR Gap Analysis."

If NRC applicants or licensees were to use MELCOR in the design or operation of nuclear power plants, the code would be subject to additional quality assurance standards for safety software. For this research analysis, the existing extensive validation is sufficient to provide high confidence in MELCOR's results.

MELCOR was used in the SOARCA project to model the reactor and plant systems, calculate thermal-hydraulics and severe accident progression, and materials, structural, and fission product behavior. MELCOR analyses also were used to both confirm the time available for operators to take mitigative actions and confirm that those measures, when successfully implemented, were adequate to prevent core damage or reduce or delay releases of radioactive material to the environment. The MELCOR modeling choices made for SOARCA were reviewed by an external peer review committee in 2006 and will be made publicly available in a MELCOR Best Practices document. The users of MELCOR in the SOARCA project were very experienced in both the code and severe accident phenomenology. To ensure technical robustness of the SOARCA process, an additional external peer review committee, which included severe accident modeling experts, reviewed the project's results and methodology.

10) Comment:

> "MELCOR does not model chemical reactions involving zirconium reacting with steam from the coolant, just with steam from the concrete."

NRC Response:

The chemical reaction mentioned on page 148 of NUREG/CR-7110, Volume 1 ($Zr + 2\ BaO \rightarrow ZrO_2 + 2\ Ba$), is only referring to the chemical effects on fission product behavior. The physical effects of energy released by this and other exothermic reactions are modeled directly in all of the MELCOR calculations. This reaction is representative of both the in-vessel damage phase and ex-vessel reaction with concrete. In both cases the release of chemical energy is included and is the dominant source of energy for fuel damage. Additional information on this topic can be found in NUREG/CR-6119 Volume 1, "MELCOR Computer Code Manuals: Primer and User's Guide, Version 1.8.5," and Volume 2, "MELCOR Computer Code Manuals: Reference Manuals, Version 1.8.5."

11) Comment:

> "Peach Bottom is a GE Mark I BWR, like 22 others in the U.S. It is a twin to the Fukushima reactors but somehow SOARCA assumes that it, and by extension other Mark I's, will not succumb to the same design flaws as their sisters in Japan. There is no basis to assume, as SOARCA does, that the vents in U.S. Mark I BWRs will operate as designed to relieve pressure buildup and prevent containment failure without important changes—making them passive and adding filters."

NRC Response:

SOARCA was an examination of important severe reactor accidents at the Peach Bottom and Surry plants. Though the Peach Bottom and Fukushima plants had many similarities, there were also important differences. Most notably is the availability of 10 CFR 50.54(hh) procedures and equipment at Peach Bottom (and all U.S. plants) designed to mitigate events that could disable large areas of a nuclear power plant. This equipment includes portable electric generators and compressed gas bottles to open containment vent isolation valves and reduce containment pressure. This equipment was only credited in the analysis of the mitigated long-term station blackout (LTSBO) scenario at Peach Bottom. Containment venting was not credited in the unmitigated LTSBO or short-term station blackout (STSBO) scenario. It is also helpful to note that in spite of the extensive damage to plant equipment that resulted from the combined earthquake and tsunami at Fukushima, plant operators were able to eventually actuate containment vent paths and control containment pressure.

12) Comment:

"How dependable is the process of using portable air bottles to open air operated valves?"

NRC Response:

SOARCA's scope did not include a human reliability analysis to calculate the probability of operators successfully using portable air bottles to open air operated valves. Instead SOARCA included two cases of each scenario—one that assumes operators are fully successful in carrying out the mitigating actions ("mitigated" case) and one that assumes the plant fails to implement 10 CFR 50.54(hh) measures and certain other actions that would prevent core damage ("unmitigated" case). SOARCA's mitigated cases of station blackouts credit the use of the portable air bottles to vent containment for Peach Bottom. No containment venting is assumed in the unmitigated cases. If containment venting is unsuccessful, this is bounded by the unmitigated cases of station blackout scenarios.

13) Comment:

"What is the present day reliability of steam driven pumps installed on old BWRs?"

NRC Response:

SOARCA did not analyze the reliability of any steam-driven pumps, including the reactor core isolation cooling (RCIC) system at Peach Bottom and the turbine driven auxiliary feedwater system at Surry. Instead SOARCA included mitigated and unmitigated cases of each scenario. For Peach Bottom, the mitigated case of the long-term station blackout (LTSBO) assumes that 10 CFR 50.54(hh) measures and procedures are successfully used and the RCIC system injects water into the core, averting core damage. For the Peach Bottom unmitigated case of the LTSBO, SOARCA assumes operators are unable to successfully implement 10 CFR 50.54(hh) equipment and procedures and therefore cannot use the RCIC system to cool the core.

14) Comment:

> "How confident are you that the black-run of the reactor core isolation cooling (RCIC) system after station battery exhaustion can continue to cool the core?"

NRC Response:

Long-term black-run of RCIC is not assumed to occur in the SOARCA calculations of the unmitigated long-term station blackout (LTSBO) sequence, nor in the short-term station blackout (STSBO) with RCIC black-start. In both cases, the absence of electric power disables instrumentation needed to indicate reactor vessel water level. Lacking this information, operators are assumed to maintain RCIC operation at a constant flow rate, which leads to over-fill of the reactor vessel and termination of RCIC by flooding the main steam lines. Although it is possible that operators could manually throttle steam flow to the RCIC turbine and reduce flow to maintain a stable reactor water level, this outcome was not judged to be sufficiently likely (given the extensive loss of support systems assumed in these scenarios) to credit long-term operation of RCIC.

Conversely, extended black-run operation of RCIC is credited in the mitigated LTSBO sequence. However, in this case, operators benefit from restoration of reactor vessel water level indications. Before direct current (dc) power from the station batteries expires at 4 hours, a portable generator is engaged to maintain a long-term supply of control power. This permits the open safety relief valve (SRV) to remain open, thereby maintaining reactor pressure at reduced levels. It also provides electric power to critical plant instrumentation, providing operators with the information they need to throttle RCIC flow, as needed, to maintain a stable water level. If, for some reason, RCIC were to trip, a portable injection pump also could be aligned to replace the lost reactor coolant injection flow. The availability of this equipment and the procedures needed to direct their operation were sufficient to examine the effects of successful mitigation.

15) Comment:

> "The ingenuity and persistence of the Fukushima Daiichi operators and their managing supervisors is seriously overlooked where 8 hour station batteries are given credit for powering some instrumentation. INPO Report 11-005, page 8 clearly states that flooding caused the 8 hour batteries to be lost. Operators were sent out to gather batteries from cars parked there at the station. After they brought them into the control room, they hooked them up and got them to work."

NRC Response:

The SOARCA analyses assumed station batteries are exhausted at 4 hours for the Peach Bottom LTSBO and 8 hours for the Surry LTSBO. These battery durations are estimates from plant system engineers at each plant. Though the technical specifications requirement for battery duration is only 2 hours, it is expected that operators will shed nonessential loads to increase battery duration. SOARCA does not make assumptions on ad hoc efforts by plant operators to gather and connect car batteries because there was no basis for this at the time SOARCA calculations were run.

Emergency Response Analysis

16) Comment:

> "SOARCA assumes that resources are available in the U.S. to mitigate a severe accident within 48 hours. This assumption is not based on real-world experience."

NRC Response:

The 48-hour truncation time for SOARCA was based on the many resources available at the State, regional, and national level that would be available to mitigate a severe reactor accident. The staff reviewed available resources and emergency plans and determined that adequate mitigation measures (at minimum, the ability to flood the reactor building) could be brought onsite within 24 hours and connected and functioning within another 24 hours. The decision to truncate releases at 48 hours (72 hours for the Surry LTSBO) was made well before the Fukushima accident. Based on the assumptions made for SOARCA, the releases that would occur within 48 hours for the Peach Bottom unmitigated scenarios cease because of reactor building flooding. For Fukushima, as discussed above, the operators delayed releases beyond the SOARCA assumption, so substantial releases occurred beyond 48 hours. In addition, the operators at Fukushima were not able to flood the reactor buildings, as assumed for SOARCA. There are significant differences between emergency response programs in the United States and Japan. The response at Fukushima does not reflect the response expected at NRC-licensed plants.

For mitigated cases, the SOARCA analysis assumed the effectiveness of mitigation measures well within 48 hours. This assumption is considered reasonable, given the vast network of resources available in the United States. These resources include an offsite emergency operations facility, which would provide access to fleetwide emergency response personnel and equipment, including the 10 CFR 50.54(hh) mitigation measures and equipment from sister plants. These assets, as well as those from neighboring utilities and State preparedness programs, could be brought to bear on the accident if needed. In addition, SOARCA did not analyze a tsunami as the initiating event, and such an event is considered highly unlikely at Peach Bottom and Surry. If sites were subject to tsunamis, these events could affect the availability and effectiveness of mitigation measures. In response to the recommendation of the NRC's Near-Term Task Force report, SECY-11-0093, dated July 12, 2011, the NRC is currently evaluating if changes to mitigation strategies are warranted.

17) Comment:

> "SOARCA emergency planning assumptions are very optimistic; they minimize risk by assuming a larger percentage of the population will be able to get out of harm's way in a timely manner."

NRC Response:

An objective of the SOARCA project was to model emergency response in a more detailed and realistic manner using current, site-specific emergency planning information. The analysis included modeling of the timing of onsite and offsite decisions and implementation of protective actions applied to multiple population segments (cohorts). Advances in consequence modeling made it easier to integrate protective action decision timing and response of the public into the consequence analysis, resulting in an evolutionary advancement over previous studies.

The assumptions used to model emergency response are based upon U.S. Federal Emergency Management Agency (FEMA) approved emergency plans that have been tested and inspected for many years. There is a good basis to assume that these well-practiced plans will be implemented as demonstrated in inspected exercises. Further, the NRC also activates its emergency response capability during accidents and the staff would be monitoring the situation to assist in decisionmaking. The NRC Chairman would be in communication with the Governor to offer assistance. These multiple levels of response capability are regularly exercised. However, the SOARCA study also conducted a sensitivity analysis to represent a delay in decisionmaking or implementation. The effect of delay was reported. It is possible that further delays or unanticipated problems could develop, but the SOARCA study is a staff best estimate of the potential consequences of the identified accidents. It is our best estimate that the emergency plans will be implemented as repeatedly demonstrated by the dedicated staff of offsite response organizations.

18) Comment:

> "Shadow evacuation was considered only out to 20 miles from the site; and only 20% of the population from 10-20 miles would choose to evacuate based on a pre-Fukushima telephone survey. Post-Fukushima, the public is likely to react differently than NRC assumed from earlier telephone samples from a small population group."

NRC Response:

Research in NUREG/CR-6864, "Identification and Analysis of Factors Affecting Emergency Evacuations," confirmed that shadow evacuations exist, but they do not typically affect the success of an evacuation. NUREG/CR-6953 Volume 2, "Review of NUREG-0654, Supplement 3, 'Criteria for Protective Action Recommendations for Severe Accidents'–Focus Groups and Telephone Survey," included a statistically relevant national telephone survey of residents of emergency planning zones with an error of plus or minus 3.5 percent at 95 percent confidence. However, the size of the shadow evacuation is very dependent on the quality of emergency messaging by local authorities. The NRC recently has published additional guidance to enhance emergency messaging in Supplement 3 to NUREG-0654. A larger shadow evacuation could take place, but with proper messaging it need not affect the evacuation times of the public within the emergency planning zone. SOARCA is a staff best estimate of the potential consequences of the identified accidents. The staff based its model on the available data for the likely size of the shadow evacuation. Other outcomes are possible, but for modeling purposes a shadow evacuation of 20 percent of the population was selected. It should be noted that the sensitivity analysis for delay in evacuation implementation also addresses the consequences of an extended evacuation time. In this manner, the effect of a larger shadow evacuation is encompassed.

19) Comment:

> "The licensee's evacuation time estimates (ETE) were used to estimate evacuation times in SOARCA. They did not take into consideration variables that would slow evacuation in reality: an extensive shadow evacuation; evacuation during inclement weather coinciding with high traffic periods; notification delay due to the fact that notification is largely based on sirens that cannot be heard indoors above normal ambient noise with windows closed or air conditioning systems operating."

NRC Response:

SOARCA's analysis of emergency response included a sensitivity analysis to assess a delay in the implementation of protective actions. This delay could occur in notification to offsite authorities, notification from offsite authorities to the public, receipt of the warning by the public, or for other reasons. Results of this sensitivity show almost identical individual annual risk of a long-term cancer fatality as the base case without the 30-minute delay.

SOARCA modeled emergency response in greater detail than any previous study. The evacuation tail noted in the study includes those members of the public that did not receive the initial alert and notification signal. This cohort is much delayed in comparison to the general public.

While the worst-case weather was not modeled, actual weather was and this likely included rain that was modeled as slowing down travel speeds. In the case of weather that significantly affects travel, it should be noted that the weather (heavy rain, high winds, heavy snow) also affects the radiological source by increasing dispersion or depositing radionuclides. As SOARCA is a staff best estimate of the potential consequences of the identified accidents, actual weather was used in the analyses.

In addition, since many of the accidents in SOARCA are assumed to be initiated by a seismic event, SOARCA includes a sensitivity case for each plant to assess how the earthquake would affect evacuation, including damage to roads and bridges, loss of traffic signals, and changes in emergency responder priorities. This is discussed in greater detail in section 6.5 of

NUREG/CR-7110, Volumes 1 and 2. The overall impact of the seismic activity on emergency response at Peach Bottom and Surry is insignificant. Prompt fatality risk remains zero for the cases analyzed.

MACCS2 and Offsite Consequence Analysis

20) Comment:

> "The SOARCA used the outdated MACCS2 computer code to analyze consequences and limited input data."

NRC Response:

The MELCOR Accident Consequence Code System, Version 2 (MACCS2) model is more than adequate to predict mean health effects risks from hypothetical severe reactor accidents. The NRC developed MACCS2 specifically as a PRA tool. It was not developed as a tool for evaluating the real-time progress of a plume or for guiding emergency response; other codes, such as Radiological Assessment System for Consequence Analysis (RASCAL), are better suited for that purpose. MACCS2 was used in SOARCA for exactly the purpose for which it was developed.

Quality assurance (QA) of the MACCS2 code has been addressed in a variety of ways. The combination of these independent QA paths has greatly enhanced the overall quality of the code. First, the implementation of the Gaussian plume model and other aspects of the modeling have been verified and documented by the developers as part of their ongoing QA effort, in compliance with requirements set by the NRC. In addition, ongoing use and independent QA efforts by many of the MACCS2 users have uncovered a few bugs and resulted in a higher level of QA than could have been obtained by the QA efforts of the developer alone. Because of the large user community of MACCS2 users, ongoing scrutiny by the users has led to valuable testing and evaluation of the code, leading to greater confidence in its quality. Finally, an assessment of the atmospheric dispersion and deposition portion of the MACCS2 code was performed and documented in NUREG/CR-6853, showing that MACCS2 compares favorably with other, higher-fidelity codes.

The adequacy of MACCS2's straight-line Gaussian plume model is discussed in a separate response. The sufficiency of using weather data from 1 year and using weather data based on site meteorological tower observations are also discussed in separate responses.

21) Comment:

> "A fundamental defect of SOARCA is that it uses a straight-line Gaussian plume model. This model does not allow consideration of the fact that the winds for a given time period may be spatially varying. A variable plume model is appropriate for reactors near large bodies of water, river valleys, and hilly terrain. SOARCA's choice of a straight-line

Gaussian plume rather than a variable trajectory model drastically reduced the size of the area that might potentially be impacted by a release of radioactive materials."

NRC Response:

The use of a straight-line Gaussian plume model is appropriate at both Peach Bottom and Surry for the SOARCA analyses. The SOARCA analyses are probabilistic in nature and are not intended to provide a highly accurate representation of a unique weather scenario, as would be required to guide emergency response for an actual accident. Previous comparisons with other, more detailed, codes have demonstrated that the Gaussian-plume model used in MACCS2 compares favorably and provides results that are generally within a factor of 2 when mean outcomes are sought (NUREG/CR-6853, "Comparison of Average Transport and Dispersion Among a Gaussian, a Two-Dimensional, and a Three-Dimensional Model"). Furthermore, this same study demonstrates that MACCS2 often outperforms the other two Gaussian plume models evaluated in NUREG/CR-6853, which account for variable wind trajectories, when compared with the highest fidelity National Atmospheric Release Advisory Center (NARAC) code. None of the conditions at either of the two sites studied as part of SOARCA would negate the conclusions of this report.

22) Comment:

"One year of meteorological data is insufficient."

NRC Response:

The SOARCA project used one year of meteorological data for the best-estimate analysis of each site. This was primarily accomplished through a cooperative effort, with the licensee using onsite meteorological tower observations. Each licensee provided 2 years of weather data. SOARCA based the specific year of data chosen for each reactor based on data recovery (greater than 99 percent being desirable) and proximity to the target year for SOARCA, which was 2005. Also, SOARCA ensured that the year's worth of weather data for each plant included statistically significant portions of each atmospheric stability type—unstable, neutral, and stable—and also statistical consideration of weather for which precipitation causes the plume to be deposited within 20 miles of the reactor site. This is discussed further in Section 5.2.1 of NUREG-1935.

SOARCA's reported offsite consequences are means of the distribution of hundreds of weather trials for the given year. It has been established from a long history of experience with severe accident mitigation alternatives (SAMA) analyses that mean consequences from 1 year of weather data differ from mean results from another year by no more than about 20 percent. So, it is very unlikely that choosing another weather year would have increased the predicted consequences by more than 20 percent. Furthermore, it is just as likely that the weather year selected for SOARCA overpredicts the health risk as it is that it underpredicts it.

23) Comment:

"Meteorological data from each plant's onsite meteorological tower is insufficient."

NRC Response:

NUREG/CR-6853, "Comparison of Average Transport and Dispersion Among a Gaussian, a Two-Dimensional, and a Three-Dimensional Model," quantifies the differences between MELCOR Accident Consequence Code System, Version 2 (MACCS2), which uses data from a single weather tower, Radiological Assessment System for Consequence Analysis (RASCAL), which uses data from multiple weather towers, and Lagrangian Operational Dispersion Integrator (LODI), which uses gridded three-dimensional data. This report demonstrates that the different methods, each using its own level of weather data, agree within a factor of 2 in terms of mean values over a year of data. Furthermore, MACCS2 did not show an overall bias when compared with the other codes. This conclusion—that differences between atmospheric transport predictions based on different codes using different models with different levels of weather data—should apply to the sites chosen for the SOARCA analyses. Thus, we should expect roughly a factor-of-2 uncertainty resulting from the atmospheric transport portion of the SOARCA analyses.

24) Comment:

"SOARCA should include other offsite consequence metrics like environmental contamination and losses of economic productivity."

NRC Response:

The objective of SOARCA was to calculate best estimates of the radiological health consequences of potential severe reactor accidents for Peach Bottom and Surry. Initially, the SOARCA project included plans to calculate early and latent cancer fatality risks and land contamination and economic consequences. However the Commission directed the staff to focus on health consequences instead of delaying the project to include an assessment of land contamination and economic consequences.

25) Comment:

"SOARCA did not model releases of contaminated water from the plant, thereby minimizing consequences. Fukushima showed the need for flooding the reactor (vessel, containment, pool) with huge amounts of water. Lessons learned for severe accidents are that enormous quantities of contaminated water are likely to enter water bodies (adding to the radioactive atmospheric fallout on the water and runoff) posing significant offsite consequences and costs, threatening the health of citizens and the ecosystem and damaging the economy."

NRC Response:

SOARCA calculates offsite consequences of severe reactor accidents in terms of individual average early fatality risk and individual average long-term cancer fatality risk. SOARCA assumes that sufficient food and water is available in the United States that the public would not eat or drink contaminated food or water. Therefore, a release of contaminated water from Peach Bottom or Surry, similar to the release at Fukushima, would be expected to have a negligible impact on the reported health risks.

26) Comment:

> "SOARCA needs to be revised and expanded to take into account the non-fatal thyroid cancers which, as Chernobyl has shown, seem likely to be the greatest observable physical health consequence of a major nuclear accident. To look only at latent cancer fatalities, when thyroid cancer is a seldom fatal, but still a serious and lifelong disease, produces a skewed and highly misleading picture."

NRC Response:

In estimating health effects from a severe accident, SOARCA calculated the radiation exposure to the population and then applied dose-response models to analyze early fatality and latent cancer fatality risks. SOARCA used latent cancer expression coefficients for the U.S. population based on BEIR V risk projection models, as detailed in the U.S Environmental Protection Agency's (EPA's) publication "Estimating Radiogenic Cancer Risks" (EPA 402-R-93-076, 1994) and implemented in EPA's Federal Guidance Report 13, "Cancer Risk Coefficients for Environmental Exposure to Radionuclides" (FGR-13). The BEIR V report used cancer mortality as a metric because, at that time, most epidemiological studies were based on cancer mortality, not cancer incidence. EPA has not yet incorporated cancer incidence data from the BEIR VII report into a revision of FGR-13, so there are no new updated cancer risk coefficients available today.

It is recognized that a large number of children and adolescents received substantial radiation doses in the thyroid after drinking milk contaminated with radioactive iodine released during the Chernobyl reactor accident. For the SOARCA study, ingestion of contaminated food and water is not considered because adequate supplies of food and water are available in the United States and can be distributed to areas affected by a reactor accident. Thus, the risk of thyroid cancer incidence from a severe reactor accident in the United States is reduced significantly when contaminated food is interdicted from public consumption. Also, the SOARCA study indicates that the amount of radioiodine released during a severe accident at a power plant located in the United States would be much smaller in comparison to the amount of radioiodine released during the Chernobyl reactor accident, which would decrease the risk of thyroid cancer incidence.

The NRC developed quantitative health objectives in terms of early fatality risk and latent cancer fatality risk and these were the metrics reported in the SOARCA study. The calculation of these metrics enables direct comparison to NRC safety goals. An objective of the SOARCA study was to update the quantification of severe reactor accident consequences found in earlier publications, specifically NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," commonly referred to as the 1982 Siting Study. The consequence metrics chosen for SOARCA enable direct comparison to this earlier study.

27) Comment:

> "Risk coefficients in SOARCA are based on old health consequence studies such as Federal Guidance Report 13, issued in 2002 and not, as it should be, on the National Academies of Science (NAS) BEIR VII report from 2005."

NRC Response:

In estimating health effects from a severe accident, SOARCA calculated the radiation exposure to the population and then applied dose-response models to analyze early fatality and latent cancer fatality risks. SOARCA used latent cancer expression coefficients for the U.S. population as detailed in the U.S Environmental Protection Agency's (USEPA's) publication "Estimating Radiogenic Cancer Risks" (EPA 402-R-93-076, 1994) and implemented in EPA's Federal Guidance Report 13, "Cancer Risk Coefficients for Environmental Exposure to Radionuclides" (FGR-13). EPA has not yet incorporated data from the BEIR VII report into a revision of FGR-13, so there are no new updated risk coefficients available today. A revision of EPA's FGR-13 is now under way, but will take many months to complete. Therefore, SOARCA is state-of-the-art in its use of published risk coefficients. It should be noted that the overall mortality estimates of FGR-13 and BEIR VII (summed over all cancer sites) differ little, so it is expected that there would be little change in the SOARCA mortality estimates if the revised FGR-13 report was available.

28) Comment:

> "By multiplying high consequence values with low probability numbers, the consequence figures appear far less startling."

NRC Response:

The releases of radioactive material calculated in SOARCA unmitigated scenarios are significantly smaller and start later than previously calculated in the SST1 case in NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," commonly referred to as the 1982 Siting Study. The Surry unmitigated ISLOCA iodine release is calculated to be 16 percent of the core inventory, but the results are more generally in the range of 0.5 percent to 2 percent for iodine and cesium for the other unmitigated scenarios analyzed. By contrast, the 1982 Siting Study SST1 case calculated an iodine release of 45 percent and a cesium release of 67 percent of the core inventory.

Each SOARCA scenario's results are reported as a scenario-specific risk so that results are put into context by weighing the frequency of occurrence against the consequences. In addition, this provides appropriate context of the scenario-specific risk in relation to other risks. In SOARCA, the offsite consequence results are expressed as the average, annual, scenario-specific risk of a latent cancer fatality for an individual within a given distance from the plant. This enables comparison to the NRC Safety Goal and to the average annual risk of dying from cancer in the United States from all causes. The NRC Safety Goal for latent cancer fatality risk from nuclear power plant operation (i.e., 2×10^{-6} or two in one million) is set 1,000 times lower than the sum of cancer fatality risks resulting from all other causes (i.e., $\sim 2 \times 10^{-3}$ or two in one thousand). The calculated cancer fatality risks from the selected, important scenarios analyzed in SOARCA are thousands of times lower than the NRC Safety Goal and millions of times lower than the general U.S. cancer fatality risk. However, such comparisons have limitations.

The NRC Safety Goal is intended to encompass all accident scenarios. SOARCA does not examine all scenarios typically considered in a PRA, even though it includes the important scenarios. In fact, any analytical technique, including PRAs, will have inherent limitations of scope and method. As a result, comparison of SOARCA's scenario-specific latent cancer fatality risks to the NRC Safety Goal is necessarily incomplete. However, it is intended to show that adding multiple scenarios' low risk results in the $\sim 10^{-10}$ range to approximate a summary risk from all scenarios, would yield a summary result that is also below the NRC Safety Goal of 2×10^{-6} or two in one million. Relative to the comparison to the U.S. average individual risk of a cancer fatality from all causes, the sources of an individual's cancer risk include a complex combination of age, genetics, lifestyle choices, and other environmental factors whereas the consequences from a severe accident at a nuclear plant are involuntary and unlikely to be experienced by most individuals.

29) Comment:

"SOARCA reports mean results instead of 95th percentile results."

NRC Response:

As stated within the report, the intent of SOARCA was to produce best-estimate calculations of the likely consequences of a severe reactor accident. To meet this objective, the mean, population-weighted individual risk was chosen to convey the likelihood of long-term cancer fatalities resulting from an accident at a nuclear power plant. This value is most meaningful in the sense that it may be compared with cancer fatality rates that have other causes. Individual risks can be presented as conditional risks (i.e., as if the accident had taken place) or as absolute risks (i.e., accounting for the likelihood of the accident occurring per year of reactor operation). The latter definition of risk is more useful, because it conveys the full meaning of risk, which is probability (or frequency) times consequence.

30) Comment:

> "SOARCA assumed that if the population moved 10 miles beyond the evacuation zone that they would be exposed to no further dose. Post Fukushima, there is no basis for that assumption."

NRC Response:

The assumption used in the SOARCA analyses, that evacuees move out to 30 miles from the plant and then receive no further dose in the immediate aftermath of the accident, is based on the belief that the evacuees would be able to move to an area that had not been affected by the plume. Taking the other viewpoint, that evacuees would move from one contaminated region to another, does not seem like a reasonable assumption, especially considering that most of the roads would continue to be passable at both sites, even for the cases in which the initiating event is an earthquake. The distance that the evacuees would need to travel before they would exit the region affected by the plume assumed in SOARCA is a reasonable one and exceeds the value of 20 miles used in previous analyses, such as in NUREG-1150.

Most of SOARCA's calculated exposure to people occurs over long periods of time after they are allowed to return to previously evacuated areas. SOARCA modeled evacuees returning home based on guidance that outlines when it would be safe to do so. For the Surry model, SOARCA uses the U.S. Environmental Protection Agency's "Manual of Protective Action Guides for Nuclear Incidents" to determine when the population can return to an area. For the Peach Bottom model, SOARCA uses Pennsylvania-specific criteria.

31) Comment:

> "The SOARCA analyses do not treat ingestion of contaminated food and water, reasoning that abundant alternatives are available in the U.S. Japan has shown otherwise."

NRC Response:

The SOARCA analyses are based on reasoning that the Nation's abundant supplies of food and water can be distributed to areas affected by a reactor accident. In addition, contaminated food and water would be interdicted from the food supply as prescribed by guidance from the U.S. Food and Drug Administration entitled, "Accidental Radioactive Contamination of Human Food and Animal Feeds: Recommendations for State and Local Agencies."

32) Comment:

> "How does the NRC safety goal compare with U.N. International Atomic Energy Agency guidelines?"

NRC Response:

The International Atomic Energy Agency (IAEA) has not issued guidelines that are directly comparable to the NRC's Safety Goals. For existing nuclear power plants, IAEA INSAG-12, "Basic Safety Principles of Nuclear Power Plants," established a target of less than 10^{-4} per plant operating year for severe core damage accidents. As shown in Appendix D of NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing," the IAEA target is compatible with the NRC's Quantitative Health Objective for individual latent cancer fatality risk. It should be noted that the IAEA target cannot be directly used to assess the SOARCA results because it does not consider the consequences of severe accidents.

<u>Public Comment Period and Timing of NUREG-1935 Release</u>

33) Comment:

> "I suggest that the comment period on NUREG-1935 and NUREG/CR-7110 Volumes 1 and 2 be extended from 30 to 90 days. Clearly this project is complex, having required NRC about four and a half years to complete in draft form. Given the scope of the documentation, its complexity, and its significance, I consider 30 days for public comments to be too abbreviated."

NRC Response:

The NRC has made many efforts to keep external stakeholders, including members of the public, informed about the SOARCA project since it began. For many years, the SOARCA team has presented updates on the project at the NRC's annual Regulatory Information Conference and has answered questions from the public. Soon after the draft version of NUREG-1935 was released for public comments on January 31, 2012, SOARCA project staff held public meetings near the two plants included in the study, Peach Bottom and Surry, to share information about the project with members of the public. In light of the fact that only one extension request was received, the NRC considers the 30-day public comment period adequate.

34) Comment:

"It is premature for the NRC to issue the draft at this time when the lessons learned from Fukushima are admittedly not yet fully understood."

NRC Response:

SOARCA is a research project that began in 2006. Essentially all SOARCA calculations were completed by March 2011, when the Fukushima accident occurred. This accident presented real information about the progression of severe accidents and many insights with potential parallels to SOARCA's analysis of station blackout scenarios at Peach Bottom, a similarly designed plant. The SOARCA team developed an appendix to NUREG-1935 that qualitatively compares and contrasts specific accident phenomena based on information available to date. Specific topics included in the appendix are operation of the RCIC system, hydrogen release and combustion, 48-hour truncation of releases in SOARCA, multiunit risk, and spent fuel pool risk. Though not all information is currently available, the Fukushima accident shows nothing that invalidates the SOARCA analyses for Peach Bottom and Surry. As additional information becomes available, the NRC will continue to review it for lessons learned and insights potentially applicable to nuclear plants in the United States. In addition, the computer codes in SOARCA will be improved in the future as they are validated against data from Fukushima when it becomes available, similar to how the current version of MELCOR was improved after being validated against accident progression data from the Three Mile Island Unit 2 accident when its data became fully available.

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11. ABSTRACT (200 words or less)

Accident phenomena and offsite consequences of severe reactor accidents have been the subjects of considerable research over the last several decades by the U.S. Nuclear Regulatory Commission (NRC). As a consequence of this research focus, analyses of severe accidents at nuclear power reactors are more detailed, integrated, and realistic than at any time in the past. A desire to leverage this capability to address conservative aspects of previous reactor accident analyses was a major motivating factor in the genesis of the State-of-the-Art Reactor Consequence Analyses (SOARCA) project. By applying modern analysis tools and techniques, the SOARCA project developed a body of knowledge regarding the realistic outcomes of select severe nuclear reactor accidents. To accomplish this objective, the SOARCA project's integrated modeling of accident progression and offsite consequences used both state-of-the-art computational analysis tools and best modeling practices drawn from the collective wisdom of the severe accident analysis community. This study has focused on providing a realistic evaluation of accident progression, source term, and offsite consequences for select scenarios for the Peach Bottom Atomic Power Station and Surry Power Station. By using the most current emergency preparedness practices and plant capabilities, as well as the best available modeling, these analyses are more realistic than past analyses. These analyses also consider mitigative measures (e.g., emergency operating procedures, severe accident management guidelines, and Title 10 to the Code of Federal Regulations (10 CFR) 50. 54(hh) measures), contributing to a more realistic evaluation.

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