ELECTRIC POWER
RESEARCH INSTITUTE

INSIGHTS ON RISK MARGINS AT NUCLEAR POWER PLANTS

A Technical Evaluation of Margins in Relation to Quantitative Health Objectives and Subsidiary Risk Goals in the United States

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Abstract

Quantitative risk criteria or goals are employed in a variety of ways in different countries in the context of risk-informed decision-making at nuclear power plants. In the U.S., quantitative objectives play an important role with respect to formal risk applications. The metrics most commonly used relate to the frequency of core damage and of large, early releases following severe accidents. In the U.S., these metrics correspond to quantitative health objectives meant to be used as surrogates for higher level safety goals. The higher-level goals were formulated such that the operation of nuclear power plants would pose no significant additional risk to an individual and the risks to society would be comparable to or less than those associated with other forms of generating electricity. Estimating risks relative to the quantitative representations of the U.S. NRC's safety goals (that is, the quantitative health objectives – QHOs) is more complex and resource-intensive than is the case for subsidiary objectives widely used in risk-informed applications in the U.S. (that is, core damage frequency and large early release frequency, CDF and LERF). Therefore, the subsidiary objectives were derived in such a way that they provide margin to the actual safety goals. This margin reflects insights from risk assessments and severe-accident analyses available at the time the subsidiary objectives were derived.

As the understanding of severe-accident phenomena has improved over the past 30 years, perspectives have evolved regarding the margins between the subsidiary objectives and the actual safety goals, and between estimates of plant risks and the safety goals. This paper explores recent developments in severe-accident analysis and risk assessment to inform and expand on these perspectives.

The objective of this paper is to survey the latest available information regarding the progression of severe accidents at nuclear power plants and the risks to the public posed by these accidents to understand how risk characterizations compare to quantitative expressions of safety goals. More specifically, the objective is to improve understanding of the margins that may exist between these quantitative expressions and surrogate or subsidiary objectives that are used extensively in risk-informed activities in the U.S. The concept of considering margins with regards to high level requirements and subsidiary quantitative goals could be more widely applicable beyond the U.S. framework. However, variations in nuclear reactor safety policy, reactor designs, extent of use of risk information in decision-making, and other aspects is outside the scope of this paper which is, therefore, currently limited to the U.S. regarding specific insights and conclusions.

Executive Summary

Quantitative risk criteria or goals are employed in a variety of ways in different countries in the context of risk-informed decision-making at nuclear power plants. In the U.S., quantitative objectives play an important role with respect to formal risk applications. The metrics most commonly used relate to the frequency of core damage and of large, early releases following severe accidents. In the U.S., the Nuclear Regulatory Commission has defined individual and societal qualitative safety goals with corresponding Quantitative Health Objectives (QHOs) for prompt fatalities (individual) and latent cancer fatalities (societal). The QHOs are expressed in terms of a percentage of the prompt and societal risk in comparison with the risk of other accidents and cancer fatalities the U.S. population are generally exposed to. Using actuarial data, the percentage risk that the Nuclear Regulatory Commission deems acceptable to meet the qualitative goals can be calculated. Subsequently, the NRC established a hierarchical structure for safety goal implementation, in which a subsidiary goal to the QHOs was established as 1×10^{-4}/year for CDF (as a surrogate for latent cancer fatality) and 1×10^{-5}/year for LERF (as a surrogate for prompt fatality). The subsidiary goals (sometimes referred to as "surrogate goals") were derived in such a way that they provide some margin to the qualitative safety goals.

As understanding of severe-accident phenomena has improved, new perspectives are available regarding the margins between both plant risks and the subsidiary objectives and the quantitative health objectives. This paper explores recent developments to inform and expand on these perspectives.

Information available from relatively recent investigations of severe accidents, including those performed by the NRC's State-of-the-Art Reactor Consequence Analysis (SOARCA) project [12, 13] and for exploring challenges to containment integrity following the 2011 accidents in Japan [20, 21], indicate that there are significant margins between the quantitative representations of the U.S. NRC's safety goals (that is, the quantitative health objectives – QHOs) and the subsidiary objectives widely used in risk-informed applications in the U.S. (that is, core damage frequency and large early release frequency, CDF and LERF).

For example, consider the margins with respect to the risk of early fatality. If a plant were to have a frequency of large early release at the subsidiary objective of 1 x 10^{-5}/year, with consequences corresponding to the most serious releases that might be anticipated (with a conditional risk of 1.5 x 10^{-6} for one of the plants studied by SOARCA), the ratio of the QHO to the computed risk is greater than 20,000. As shown in the SOARCA study, this is primarily because releases large enough to produce early health effects and rapid enough to occur before the population adjacent to the site has been evacuated are extremely unlikely. One implication of these findings is that LERF may not be a very meaningful risk metric for risk informed decision-making. A plant could have a substantially higher frequency of large releases but still not pose a risk of early fatalities to individuals within 1 mile of the site boundary, (although it is still important from a defense-in-depth perspective to take appropriate actions to ensure that containment integrity is maintained).

These more recent analyses also provide a perspective on the margins available with respect to the risk of latent-cancer fatality. Using bounding assumptions (i.e., assuming the frequency of the severe accidents considered in these studies corresponds to the frequency of core damage at the subsidiary objective of 1 x 10^{-4}/year), the margin to the QHO is on the order of a factor of 70 to 300 (where the lower value corresponds to the overly conservative assumption that the entire core damage profile resulted in a release comparable to the more severe accident evaluated by SOARCA for one of the plants studied, i.e., 3 x 10^{-4}). It should be noted that the values considered in this paper are specific to individual plants and for certain scenarios studied under various efforts. The intent is to consider the general implication of the type of quantitative margins that more recent assessments can provide based on the technical information used.

The safety goals and subsidiary objectives were established with the intent that they would be applied relative to mean estimates of risk. Many of the results presented here are bounding in the sense that the frequencies were assumed to be at the subsidiary objectives (LERF of 1 x 10^{-5}/year; CDF of 1 x 10^{-4}/year). If the accident frequencies for a particular plant were calculated to be at these values, the vast majority of the accident scenarios that contributed to those frequencies would almost certainly be less severe than those represented by the releases assumed in calculating the consequences. This would suggest that the margins presented in this white paper may be underestimated for plants in general.

Furthermore, the margins provide an opportunity to consider the role of uncertainty in decision-making that takes into account risk information. The need to consider uncertainties has been discussed extensively, including in a relatively recent update to NUREG-1855 [25] and in companion EPRI documents [26, 27]. It is also addressed in the context of decision-making when it is necessary to aggregate various risk contributors [28].

Much of this guidance, and particularly that in NUREG-1855 [25], is focused on the need to understand uncertainties when the quantitative risk results might have the potential to lead to one or more of the subsidiary objectives being exceeded. From the information presented here, such as the uncertainty results in NUREG-1150 [5] and the more recent studies, the insights are that only in the most extreme cases could uncertainties play a significant enough role in this respect, at least with regard to cases in which the CDF approaches (or even exceeds) the subsidiary objective.

As the use of risk information continues to expand, an updated perspective on safety goals and related objectives may be of value in helping to ensure that decisions are made in an objective and effective manner.

Introduction

Quantitative risk criteria or goals are employed in a variety of ways in different countries in the context of risk-informed decision-making at nuclear power plants. In the U.S., quantitative objectives play an important role with respect to formal risk applications. The metrics most commonly used relate to the frequency of core damage

and of large, early releases following severe accidents. In the U.S., these correspond to subsidiary objectives that are meant to be used as surrogates for the actual safety goals, expressed in terms of quantitative health objectives for which risk calculations would be more complex and uncertain. The subsidiary objectives were derived in such a way that they provide some margin to the actual safety goals.

As understanding of severe-accident phenomena has improved, perspectives have evolved regarding the margins between both plant risks and the subsidiary objectives and the quantitative health objectives. This paper explores recent developments to inform and expand on these perspectives.

The question of "how safe is safe enough?" has been raised in the context of many technologies and industries. Relative to the operation of nuclear power plants, this question has traditionally been answered – indirectly – by formulating a comprehensive set of conservative deterministic rules and guidelines that, if implemented effectively, are meant to provide reasonable assurance of adequate protection.

Over the past three decades, probabilistic risk assessments (PRAs) have grown in use as an adjunct to traditional safety analyses. PRAs have proven very valuable as systematic tools for identifying residual weaknesses in plant design and operations, and in providing perspective on the relative importance of challenges to plant safety. The emergence of PRAs allowed for the development of safety goals that could be used as objective measures for judging "how safe is safe enough." Safety goals have been formulated and applied in a variety of ways in different countries.

An Example: Formulation of Safety Goals in the U.S.

In the U.S., where the most extensive formal use of risk information has been made in regulatory interactions, the Nuclear Regulatory Commission (NRC) established its Safety Goal Policy in 1986 [1]. In that policy, the NRC set forth two qualitative safety goals and corresponding quantitative health objectives (QHOs). The higher-level qualitative goals were defined as follows: the operation of nuclear power plants would pose no significant additional risk to an individual, and the risks to society would be comparable to or less than those associated with other forms of generating electricity. The associated QHOs, summarized in Table 1, were formulated such that reactor accidents should not exceed one-tenth of one percent in the risks of early fatality for individuals living adjacent to a nuclear power plant and for fatality due to latent cancer among the population in the general vicinity of a plant. Once defined in this manner, the average risks to individuals and to the population at large could be used to derive objectives against which the risks associated with a nuclear power plant could be compared using quantitative results from a PRA. These derived objectives are included in Table 1 as the derived quantitative health objectives and the subsidiary objectives.

Table 1 – Summary of U.S. Safety Goals and Objectives

Prompt Fatality Risks	Latent Cancer Risks
Qualitative Safety Goals	
Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.	Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.
Quantitative Health Objectives	
The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.	The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.
Derived Quantitative Health Objectives	
The risk of early fatality for an individual living within 1 mile of the site boundary should not exceed 5×10^{-7} per year (based on average risk of fatality due to all types of accidents of approximately 5×10^{-4} per year in the U.S.).	The risk of latent cancer fatality to an individual living within 10 miles of a nuclear power plant should not exceed 2×10^{-6} per year (based on an overall average risk of latent cancer fatality of approximately 2×10^{-3} per year in the U.S.).
Subsidiary Objectives	
The frequency of a large, early release should not exceed 1×10^{-5} per year [limiting the potential for an accident that could result in early fatalities].	The frequency of core damage should not exceed 1×10^{-4} per year [limiting the potential for an accident that could affect the population over a longer period of time].

Derivation of Subsidiary Objectives

The estimation of risk for a nuclear power plant includes the need to address the nature and impact of uncertainties. Sources of uncertainty are relevant for each stage of a PRA, and additional sources can be introduced as a PRA progresses from assessing the frequency of core damage (a "Level 1" PRA) to evaluating the frequencies and severities of accidental releases due to a core-damage accident (a "Level 2" PRA), and finally to estimating the risk to the public (that is, the frequencies of various health consequences, a "Level 3" PRA). Because of these uncertainties and the efforts required to perform full Level 2 and Level 3 PRAs, the NRC established two subsidiary objectives to facilitate decision-making in risk informed applications. These relate to the frequency of core damage and the frequency of a large early release due to a core-damage accident [2, 3].

The large early release frequency (LERF) is a surrogate for the quantitative objective related to the risk of early fatalities. An "early" release is one for which there might not be sufficient warning time to ensure effective evacuation of the population near the plant. The potential for early health effects are drastically reduced, and potentially eliminated, when the population has been evacuated. The release would be considered "large" if radioactive fission products were not reduced significantly by scrubbing or other removal mechanisms within the reactor systems or the containment building.

Similarly, the core-damage frequency (CDF) has been used as a surrogate for latent health effects [2]. If the risk of core damage is small, the risk of any releases that could result in latent effects would be limited. As discussed in several NRC documents [2, 3, 4], it was recognized during the development and adoption of the subsidiary safety objectives that these thresholds were not intended to be treated as licensing standards or requirements. These documents suggested that, due to the lack of knowledge on severe-accident phenomenology at the time the policy on reactor safety goals was issued, these thresholds could be subject to revisions when improvements in the state-of-art advanced sufficiently. As discussed in the Staff Requirements Memorandum related to SECY- 89-102 [3], the review of existing plant PRA models during the 1980s indicated that plants not only met the quantitative health effects objectives "but exceed them".

Prior State of Knowledge

Limited PRA results were available at the time the subsidiary safety objectives were established via SECY- 89-102. The first significant effort since the NRC's Safety Goal Policy [1] was published, was the study published in NUREG-1150 [5], which summarized severe accident risk assessments of five commercial nuclear power plants in the U.S. [5], from core damage frequency to offsite consequences. Three of the plants considered in NUREG-1150 (Peach Bottom, Surry, and Sequoyah) were the subject of more current studies While first issued in 1987, its final publication was completed at the end of 1990. NUREG-1150 produced point estimates of the subsidiary safety goals (CDF and LERF), and uncertainty distributions for each plant, as well as offsite radiological consequence estimates. All plants, except for Zion (no longer operating) had mean values below the subsidiary objectives and met the QHOs with significant margins. For example, for Surry, the mean individual early fatality risk would have to increase by a factor of about 30 in order to approach the early health objective; and, for Sequoyah or Zion, the individual latent cancer fatality risk would have to increase by more than two orders of magnitude to approach the corresponding health objective [6]. Although not all plants in NUREG-1150 included external event contributors, two plants did.

In 1988, the NRC issued Generic Letter (GL) 88-20 [7], requesting that all U.S. licensees perform an Individual Plant Examination (IPE) "to identify any plant-specific vulnerabilities to severe accidents and report the results to the Commission." In response to this request, submittals covering all the nuclear power plants in operation in the U.S. at the time used PRA modeling results that included internal events and internal flooding scenarios (a posterior effort covered other contributors).

A summary of the insights from the submittals to GL 88-20, published in 1997 as NUREG-1560 [6], provided a high-level assessment of the margins to the safety goals using the IPE results (most IPEs did not calculate offsite health effects as in NUREG-1150). The IPE CDFs for all BWRs and most PWRs were below the subsidiary safety objective of 1E-4/year; less than 10% of all PWRs exceeded this threshold. Similarly, while most PWRs met the LERF subsidiary safety objective, a few exceeded it. Several BWRs exceeded the threshold for early containment failure. As such, NUREG-1560 performed additional investigations using consequence results

from NUREG-1150 as a basis to evaluate what these exceedances implied. Using simplistic assumptions to produce a rough consequence of offsite radiological consequences and the NUREG-1150 results for Surry as comparison, NUREG-1560 identified a subset of plants with results that could exceed the QHOs. Further examination of this subset indicated that conservative assumptions made in the IPE submittals regarding CDF contributors and severe-accident progression accounted for the few exceptions in which the individual early fatality risk level could approach the QHO. NUREG-1560 ultimately concluded that even those IPE results that exceeded the subsidiary safety goals implied risk levels below the individual latent cancer fatality health objective, and most plants were below the individual early fatality health objective. Studies of site-specific characteristics for early impacts as a function of population were suggested as well.

A later demonstration that the subsidiary objectives are acceptable surrogates for the latent and early quantitative health objectives for operating reactors was provided in NUREG-1860 [8]. This demonstration, summarized in Appendix D of NUREG-1860, drew upon the risk analyses performed by NRC in the late 1980s for in NUREG-1150 [5]. Those risk analyses reflected the state of the art at the time they were performed. The demonstration related to LERF accounted for the following:

- The consequences were taken to be the mean conditional risk of early fatality corresponding to the more severe release categories for the Surry nuclear plant in Virginia [9]: approximately 0.03
- The frequency was assumed to be that of the surrogate objective for LERF: 1 x 10^{-5}/year

The product of these values yields a risk metric of 3 x 10^{-7}/year, which is approximately half the subsidiary QHO (5 x 10^{-7}/year).

It is important to note that this is a bounding assessment. The mean frequency for the release categories near the assumed level of consequences for Surry was approximately 4 x 10^{-7}/year [9], or nearly two orders of magnitude lower than the surrogate QHO. Assuming that this release category is representative of all of those that would contribute to LERF is, therefore, likely to be bounding by a wide margin. Furthermore, as shown in Figure 1, the value calculated in NUREG-1860 is higher than the mean and 95th percentile risk values for Surry [8].

An analogous calculation was reported in NUREG-1860 (also using the Surry results in NUREG-1150 as a basis) to confirm the acceptability of using a CDF objective of 1 x 10^{-4}/year as a surrogate or subsidiary objective for the risk of latent-cancer fatality for individuals in the vicinity of a nuclear power plant:

- The consequences reflected the largest mean conditional risk of latent-cancer fatality: approximately 4 x 10^{-3}
- The frequency was assumed to be that of the surrogate objective for CDF: 1 x 10^{-4}/year

The product of these values yields a risk metric of 4 x 10^{-7}/year, which is approximately a factor of five lower than the corresponding subsidiary QHO (2 x 10^{-6}/year), as shown in Figure 2. Similar to the case for the risk of early fatalities, the value calculated in NUREG-1860 implicitly includes additional margin because most of the frequency of core damage produces smaller conditional risks.

Figures 1 and 2 also include the uncertainty distributions for the Surry results from NUREG-1150. From a practical standpoint, it is important to note that the mean values of risk assessments are nearly always skewed toward the upper end of the uncertainty bands as a result of the types of distributions used to represent the various reliability parameters. As the uncertainties increase, the mean value tends to move upward as well (and may, in fact, exceed the 95th percentile value)

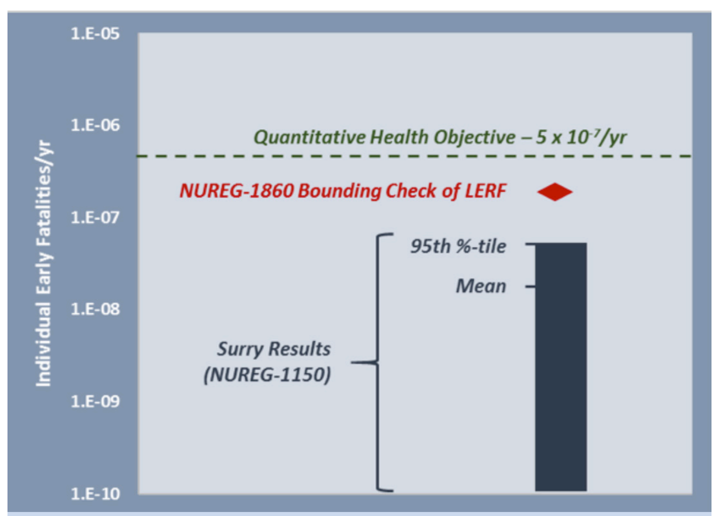

Figure 1 – Perspective on Derivation of Subsidiary QHO for Risk of Early Fatalities

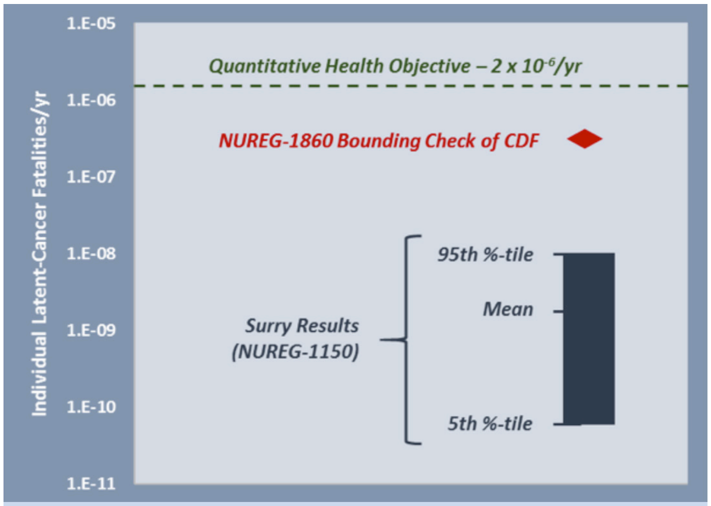

Figure 2 – Perspective on Derivation of Subsidiary QHO for Risk of Latent-Cancer Fatalities

Use of the Subsidiary Objectives in Risk-Informed Applications

While these two subsidiary objectives were never meant to serve as absolute limits for an individual nuclear power plant in the U.S., they have played a prominent role in decisions involving risk at U.S. nuclear power plants, including those stemming from the application of risk-informed decision-making as described in NRC Regulatory Guide 1.174 [10]. In particular, this becomes an important aspect of decision-making if approaching or exceeding these thresholds needs to be accounted for in formal risk-informed applications.

A clearer understanding of the margins and the implications of quantitative PRA results may prove useful in working toward expanding the use of risk information in decision-making.

Current State of Knowledge

The understanding of challenges to the safe operation of nuclear power plants has evolved substantially since the time of the first major PRA study, the Reactor Safety Study [11], and the 1979 core-damage accident at the Three-Mile Island nuclear power plant.

Some of that evolutionary understanding was captured in NUREG-1150 (which was published in 1990). Since the completion of the NUREG-1150 PRAs, understanding of the behavior of severe core-damage accidents and the potential for accidental releases to cause offsite health consequences has continued to evolve.

This evolving information offers insight into the embedded margins between the subsidiary objectives and the QHOs, as well as between estimates of risk for actual plants and the QHOs (especially if the risk profile approaches or exceeds the subsidiary quantitative goals).

More specifically, significant recent information is available from studies undertaken by the NRC to perform State-of-the-Art Reactor Consequence Analyses (SOARCA) [12, 13] and from other analyses performed in the aftermath of the 2011 Fukushima Dai-ichi accidents in Japan.

Trends in Nuclear Power Plant Risks

A white paper published by the Electric Power Research Institute (EPRI) in 2008 described the concurrent improvements in safety (as indicated by plant-specific calculations of CDF) and in overall plant performance (as indicated by increasing capacity factors) at U.S. nuclear power plants [14]. That white paper described how risk-informed activities can sharpen focus on the most important safety aspects of plant operation, enabling plants to make changes that have reduced the frequency of plant challenges (which has also contributed to the increases in capacity factors); improved the reliability of equipment important to safety; and supported the identification and elimination of weaknesses in system design and operating practices. As illustrated in Figure 3, a more recent compilation of the data addressed in the earlier white paper indicates that the trends in plant improvements have continued in the years since that paper was published. These results highlight that earlier estimates of CDF and LERF in studies such as NUREG-1150 and the IPEs as summarized in NUREG-1560 have been further refined.

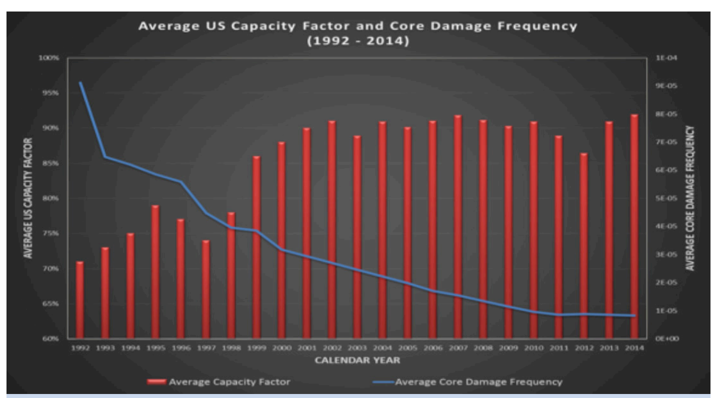

Figure 3 – Trends in Plant Capacity Factors and Core-Damage Frequencies

It should be noted that the values of CDF reflected in Figure 3 represent contributions only from internal initiating events. More recent detailed investigations of other contributors to risk, including those due to fires and earthquakes, have also resulted in identifying opportunities for further plant improvements.

State-of-the-Art Reactor Consequence Analyses

The SOARCA studies were performed for three representative plants:

- A PWR with a large, dry containment (Surry),
- A BWR with a Mark I containment (Peach Bottom), and
- A PWR with an ice-condenser containment (Sequoyah).

The SOARCA studies were not full risk assessments, since they did not attempt to explore a complete range of potential accidents. Instead, they selected a representative set of the more challenging severe accidents and focused on detailed assessment of the physical progression for these accidents, the range of radiological releases that might result, and the corresponding potential for offsite health effects. Thus, while they provide some insights into absolute estimates of risk (because they do account for accidents likely to contribute most to overall risk), the results that are more useful and relevant in understanding margins in safety goals are those that represent the conditional likelihood and severity of offsite consequences given occurrence of the more challenging severe accidents.

The SOARCA studies for Peach Bottom and Surry differ in focus from those for Sequoyah. For Peach Bottom and Surry, the initial focus was on a best-estimate investigation of the accident response and resulting consequences, followed by an assessment of uncertainties. Because Sequoyah uses an ice-condenser containment, which is smaller and has a lower design pressure than a PWR containment such as that at Surry, the primary area of interest was to address the historical concern regarding the potential for hydrogen accumulation and ignition. The Sequoyah studies were performed using an integrated approach to considering uncertainties throughout the process for the representative severe accidents considered.

Among the aspects of accident response considered by the SOARCA studies were the mitigating features incorporated as part of the U.S. approach for adding diverse and flexible mitigation strategies (also known as FLEX) [15]. Although there was no attempt to quantify the difference in frequencies of the core-damage sequences when crediting FLEX, the studies did find that FLEX could avert core damage for SOARCA scenarios investigated except those involving a short-term station blackout (that is, a total loss of ac power, with no core cooling available from the outset).

As noted above, much of the focus of the SOARCA analyses was on the progression of core-damage accidents. Detailed analyses were performed using advanced computer codes to understand potential variations in accident response and the response of plant systems [11,12,13]. Among the more significant findings of these analyses was that core damage and radiological releases were likely to be delayed substantially relative to predictions from earlier analyses. These delays are indicative of the manner in which the core degraded and the interactions of degraded core material with the reactor pressure vessel (RPV). Most earlier analyses, which often relied on bounding assessments and simplified treatments, were found to have assumed much more rapid accident progression.

This updated understanding of the rate of accident progression is significant with respect to the potential for offsite consequences because the longer times for the core to melt and for the reactor vessel to be breached result in

- More time for the population near the plant site to be evacuated effectively.
- More time for fission products to plate out on the cooler surfaces in the reactor coolant system, and therefore smaller fractions released following any subsequent containment failure.

The SOARCA studies also took advantage of further research regarding the phenomena that could lead to a large early release (what is commonly referred to as a "LERF" scenario). In the SOARCA studies for Peach Bottom and Surry, no outcomes (even for station blackout resulting from a large earthquake) resulted in large, early releases. This was largely a consequence of the timing considerations noted above and the conclusion that phenomena previously considered to present important challenges to containment integrity at the time the core debris exited the reactor vessel (including direct containment heating[1]) were very unlikely. The Sequoyah studies similarly found that early containment failure due to combustion of hydrogen generated during core degradation could only occur under certain unlikely sets of conditions.

1 Direct containment heating refers to a set of phenomena following ejection of molten core material from the reactor vessel at high pressure. Fine particles of very hot molten core material were thought to have the potential to transfer energy to the containment atmosphere at a very rapid rate, resulting in further generation of hydrogen that could burn (via oxidization of the molten core material). This was expected to add further to the heatup and pressurization of the containment (a very important contributor to the potential for early containment failure in the NUREG-1150 studies).

Another potential source of a large, early release for PWRs has traditionally been a steam generator tube rupture scenario. Accidents following an initial steam generator tube rupture, however, typically evolve at a slow rate. A more significant concern was the potential for a tube rupture to be induced by hot gases transported to the steam generators while the core was melting. Through detailed modeling of the conditions in the reactor coolant system (RCS), the SOARCA analysis for Surry concluded that a more likely outcome was for a thermal failure to occur in an RCS hot leg, rather than by failure of a steam generator tube. If that were to occur, the stresses on the steam-generator tubes would be alleviated, and no tube failures would be expected. Even if a tube were to fail before an induced failure in a hot leg, the release would be delayed for many hours if feedwater were available.

Another aspect investigated in the SOARCA studies was the extent to which fission products might be released to the environment in the event of an accident – the *source term*. When addressing accident source terms, iodine and cesium are often used as representative radionuclides with short (iodine) and long (cesium) half-lives. The SOARCA studies concluded that source terms would generally involve smaller release fractions than in previous assessments:

- Releases of iodine were typically limited to 1 to 2% of the core inventory for the dominant accidents, and 10 to 15% for even more severe (and less frequent) accidents.
- Releases of cesium were predicted to be 2% or less of the total in the reactor core.

In addition to updated treatment of the accident progression and lower estimates of fission-product releases than had been assumed in previous studies, the SOARCA studies explored what were considered to be more realistic simulations of emergency response. SOARCA considered in detail potential evacuation routes and accounted for changes in evacuation speeds based on the extent of traffic congestion and accounted for reduced speeds when the (hourly) meteorological data included precipitation. These factors combined to result in SOARCA predicting "essentially zero" individual risk of early fatalities for any of the three plants. Only for particular sensitivity cases (e.g., with less effective evacuation or with even more unlikely assumptions regarding earlier containment failure, for example) were any early fatalities calculated to occur.

Because of the effectiveness of the assessed evacuation of the population out to 10 miles (16.1 kilometers) from the plant, the population would not be expected to be exposed to high doses. Therefore, the risk of fatality due to latent cancers comes from exposure to relatively low doses, much of it assumed to be received by persons returning to their homes after having been evacuated. This dose would be controllable to some extent (that is, people could be prevented from returning to their homes if the dose rate remained above some level, as has been the case in the area around the Fukushima Dai-ichi plant in Japan).

The level of latent cancer risk is also affected by whether or not a threshold (or dose truncation) is applied. Risk analyses have traditionally employed the "linear no-threshold" (LNT) assumption. That is, the risk of latent cancer fatality is proportional to the dose received; no matter how small a dose was received, it would have a finite potential to cause a fatality. The LNT assumption has been a subject of debate [16, 17, 18, 19], i.e., while its simplicity makes it a convenient model for use in regulatory interactions, it may have limited scientific validity.

To examine the LNT assumption, the SOARCA studies reported the results for two sensitivity studies. One was based on a position paper developed by the Health Physics Society, which recommended not estimating risks below a certain amount above background radiation [16] (among other references such as [17,18,19]). The other sensitivity case considered truncated doses at background levels in calculating the risk of latent cancer fatality. As summarized in Table 2, these sensitivity studies predicted latent cancer fatalities at a rate one to two orders of magnitude lower than was calculated using the LNT assumption. The results of these sensitivity studies suggest that further research may change the characterization of latent consequences in future risk assessments.

It should be noted that the level of detail and in-depth assessment in the SOARCA studies regarding severe accident progression well exceeded the simplifications used in NUREG-1150 and the IPEs as summarized in NUREG-1560, highlighting areas where prior studies overestimated the impacts, particularly with respect to early health effects.

Table 2 – Sensitivity of Latent-Cancer Risk to Assumptions on Dose Threshold[1]

Scenario	Conditional Probability of Latent Cancer Fatality by Scenario[2]		
	Linear No-Threshold (Base case)	**Background**	**Health Physics Society**
Peach Bottom			
Long-term station blackout	9 X 10^{-5}	7 X 10^{-7}	3 X 10^{-7}
Short-term station blackout with RCIC blackstart[3]	7 X 10^{-5}	7 X 10^{-7}	3 X 10^{-7}
Short-term station blackout without RCIC blackstart	2 X 10^{-4}	1 X 10^{-5}	1 X 10^{-5}
Surry			
Long-term station blackout	5 X 10^{-5}	3 X 10^{-7}	1 X 10^{-9}
Short-term station blackout	9 X 10^{-5}	3 X 10^{-6}	1 X 10^{-8}
Short-term station blackout with thermally-induced rupture of steam generator tube	3 X 10^{-4}	8 X 10^{-5}	1 X 10^{-5}
Interfacing-systems loss-of-coolant accident	3 X 10^{-4}	7 X 10^{-5}	3 X 10^{-5}

Notes
1. Entries in this table are consolidated from Tables 6 through 9 of NUREG-1935 [12].
2. Values represent probability of latent-cancer fatality for an individual located within 10 miles of the plant, conditional on occurrence of the scenario.
3. "RCIC blackstart" refers to the ability to start the reactor core isolation cooling (RCIC) pump without ac or dc power.

The SOARCA studies provided many conclusions related to the safety of nuclear power plants:

- "The individual early fatality risk from SOARCA scenarios is essentially zero." It is important to note that the scenarios were selected for investigation based on the severity of their challenges to the plant (i.e., station blackout, interfacing-systems LOCA, etc.).
- "Individual LCF [latent cancer fatality] risk from the selected specific, important scenarios is thousands of times lower than the NRC Safety Goal and millions of times lower than the general cancer fatality risk in the United States from all causes, even assuming the LNT dose-response model."
- "SOARCA results indicate that bypass events (e.g., Surry ISLOCA) do not pose a higher scenario-specific latent cancer fatality risk than non-bypass events (e.g., Surry SBO). While consequences are greater when the bypass scenario happens, this is offset by the scenario being less likely to happen."
- "SOARCA reinforces the importance of external events relative to internal events and the need to continue ongoing work related to external events risk assessment."
- "The SOARCA analyses show that emergency response programs, implemented as planned and practiced, reduce the scenario-specific risk of health consequences among the public during a severe reactor accident. Sensitivity analyses of seismic impacts on site-specific emergency response (e.g., loss of bridges, traffic signals, and delayed notification) at Peach Bottom and Surry do not significantly affect LCF risk."
- "SOARCA results, while specific to Peach Bottom and Surry, may be generally applicable for plants with similar designs. However, additional work is needed to confirm this, since differences exist in plant-specific designs, procedures, and emergency response."

Post-Fukushima Assessments

A separate set of analyses undertaken by the U.S. NRC [20] and by EPRI [21] in the aftermath of the 2011 accidents at Fukushima Dai-ichi provide additional perspective on the risk of severe accidents for a nuclear power plant. These analyses were performed to address "Containment Protection and Release Reduction" (CPRR) for BWRs (particularly those with Mark I containments)

- To gain further insight into the challenges potentially posed by severe accidents,
- To identify options for more effective management of such accidents, and,
- In the case of the NRC analyses, to provide a technical basis for further potential regulatory actions.

The basic configuration of a BWR Mark I containment is shown in Figure 4; this figure may be useful in understanding the discussion that follows.

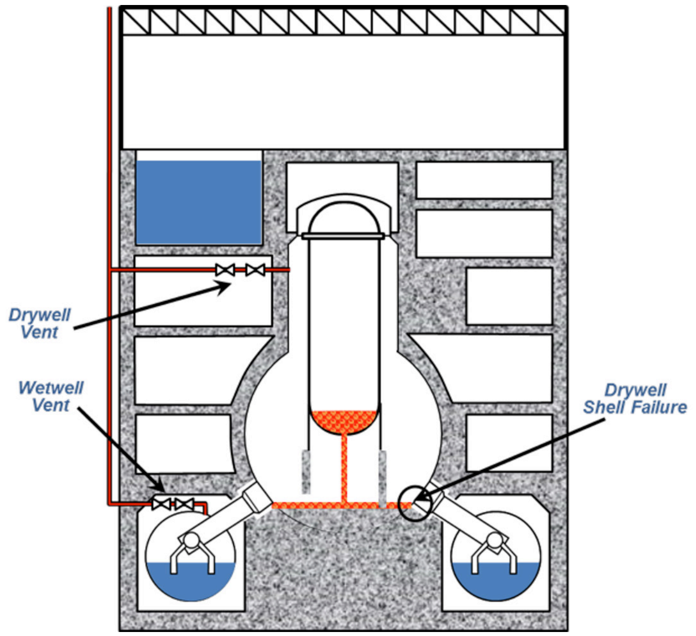

Figure 4 – Simplified Configuration of a BWR Mark I Containment

Both the NRC and the EPRI studies focused on accidents involving an extended loss of ac power (ELAP). They evaluated a variety of potential strategies for managing plant response during such an accident to understand their effectiveness in preventing containment failure and in limiting potential radionuclide releases. The EPRI study considered 24 alternatives that were comprised of various combinations of

- Containment vent configurations, including addition of a drywell vent and reliance on manual actions vs. passive means to initiate venting
- Means to add (and manage) water to the RPV or to the drywell
- Small engineered severe-accident filters
- Large engineered filters
- Large engineered filters with a rupture disc on the vent line from containment.

The risk of latent-cancer fatalities for persons within 10 miles (16.1 kilometers) of the site was calculated for all of the alternatives considered. The results are shown in Figure 5.

As Figure 5 shows, even the upper end of the uncertainty ranges for the base case (i.e., Alternative 1, accounting for no further action to manage severe accidents) is more than two orders of magnitude below the QHO. It should be borne in mind that these results do not reflect the full core-damage spectrum, since they considered only ELAP scenarios (initiated by internal events or seismic events). Nevertheless, these ELAP scenarios would be expected to be among the most challenging accidents; the results, therefore, give insight into the margins available to the QHO.

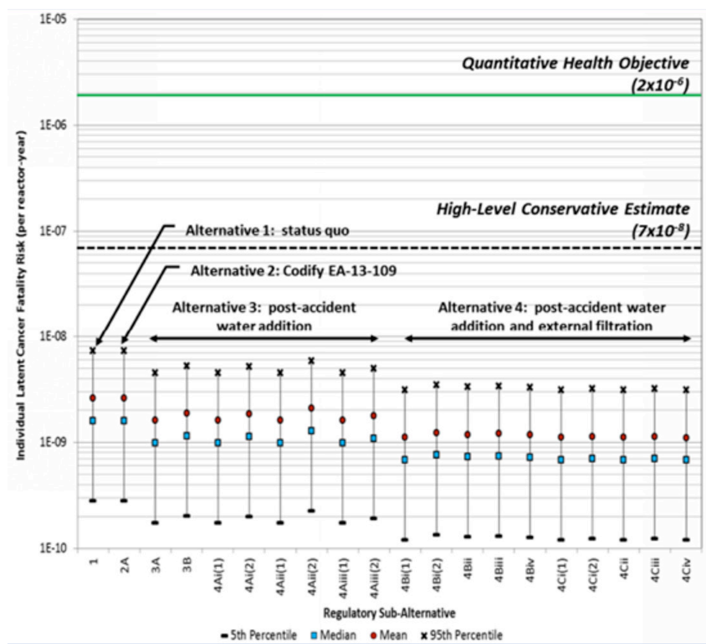

Figure 5 – Individual Latent-Cancer Fatality Risk, with Uncertainty Bounds for Alternatives (Figure 3-3 from the NRC CPRR Study [20])

The EPRI study paralleled in many respects that carried out by the NRC. EPRI used its own code to explore the physical progression of the accident scenarios (i.e., MAAP), but used the same code as the NRC in the SOARCA studies for the calculation of offsite consequences (i.e., MACCS).

In the base-case assessment, the EPRI analyses found that an extended station-blackout accident, with no additional capabilities for adding water or venting containment, would be very likely to result in a failure of containment shortly after the core melted through the RPV. This failure would be the result of hot core debris spreading across the drywell floor and coming into contact with the steel shell of the drywell. A failure of the drywell shell at that point could create a release path through the reactor building and to the atmosphere. The risk of latent-cancer fatality was calculated for this base case to be approximately 3.7×10^{-9}/year (only slightly higher than the analogous case from the NRC study). While this frequency accounts only for the ELAP scenarios (with an estimated core-damage frequency of 7.3×10^{-6}/year), most other core-damage scenarios would present less severe challenges to containment integrity based on scenarios typically modeled in a PRA (i.e., albeit not quantified here, it is not expected that all scenarios will have the same level of consequence as ELAP scenarios).

As noted earlier, 24 cases representing different potential changes to the plant configuration were evaluated. The results with respect to the conditional probability of a containment failure during an ELAP are shown in Figure 6 for all of the cases.

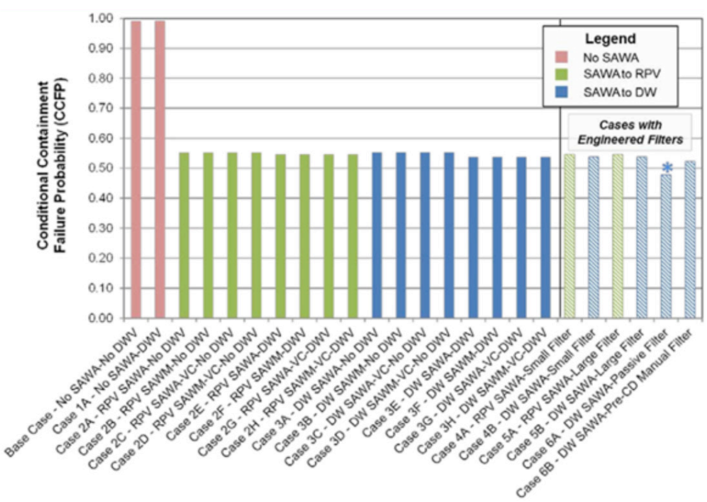

Figure 6 – Conditional Probability of Containment Failure for Severe-Accident Management Alternatives (Figure 5-1 from EPRI 3002003301 [21])

The NRC assessment considered four basic options with respect to containment vent configurations, means to add water to the RPV or to the drywell, and combinations of engineered filters, with some variations, but addressed largely the same considerations as those in the EPRI study.

The NRC study used the MELCOR code to evaluate the severe-accident response of the plant and to calculate the source terms for release (MELCOR is a code developed and maintained by the Sandia National Laboratories for the NRC). The MACCS code was used to calculate offsite consequences. The NRC study found that the risk of individual early fatality was zero because the source terms were not large enough to exceed the threshold for an acute dose. This remained true for a variety of sensitivity studies that examined cases such as population density near the plant, delay in initiating evacuation in the area surrounding the plant, and assumptions regarding the number of people who did not evacuate.

As Figure 6 illustrates, one of the elements essential to reducing the potential for containment failure is to provide for water injection during a severe accident. If sufficient water is successfully injected into either the RPV or directly into the drywell, core debris may be quenched as it exits the RPV, and direct attack on the drywell shell can be averted. Providing for severe-accident water addition (SAWA), therefore, reduced the conditional probability of containment failure from near certainty (about 99%) to about 55%. The remaining probability of containment failure results from the potential that SAWA may not be successful, or from other modes of containment failure. As can be seen in Figure 6, this conditional probability is essentially the same for all of the alternatives.

The individual risk of latent-cancer fatalities is presented in Figure 7 for the same 24 alternatives. Even for the base case, the risk is nearly three orders of magnitude lower than the applicable QHO. A modest reduction is realized for all of the alternatives, due to the provisions for severe-accident water addition. This action both reduces the potential for relatively early containment failures and increases the extent of fission-product scrubbing. As Figure 7 illustrates, the six alternative cases with engineered filters provide only a small reduction in risk compared to the other alternatives.

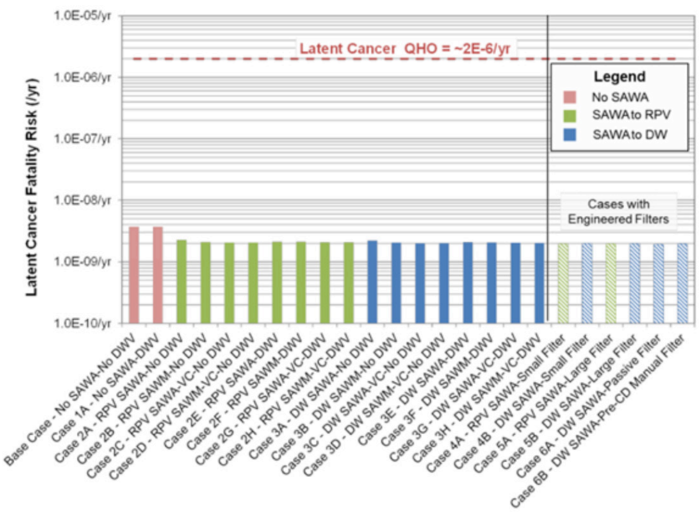

Figure 7 – Mean Risk of Latent-Cancer Fatalities for Severe-Accident Management Alternatives (Figure 5-2 from EPRI 3002003301 [21])

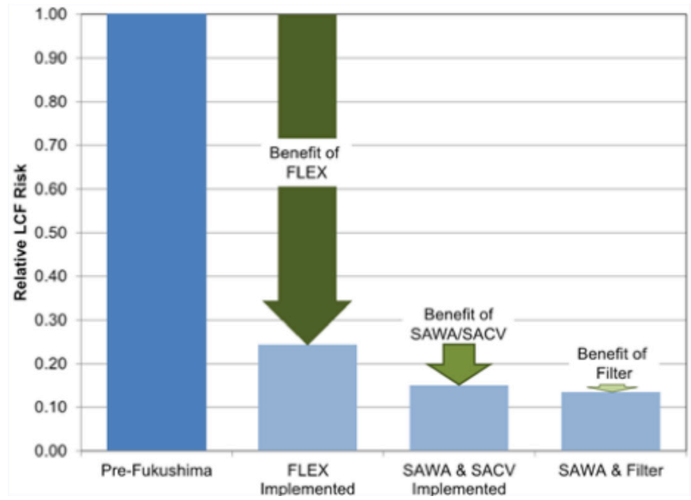

Figure 8 – Relative Benefits of Potential Post-Fukushima Enhancements (Figure 5-3 from EPRI 3002003301 [21])

Figure 8 compares the relative risks of U.S. plant configurations prior to the Fukushima accidents to the impact with implementation of FLEX, and to the impact associated with severe-accident water addition and a severe-accident capable vent (referred to as "SACV" in Figure 8). Once again, the additional benefit of an engineered filter is shown to be small. This small impact is the result of two important aspects of the potential severe accidents:

- For the fraction of ELAP scenarios that still result in a failure of containment, the major releases would bypass the filtered vent.
- For other accidents, releases through the vent systems would be scrubbed (especially for those releases that took place through the vent line on the suppression pool, or wetwell). The water in the suppression pool or water overlying core debris in the drywell would accomplish much of the removal of fission products that would be achieved using an engineered filter.

Several sensitivity studies were performed to gain further insight into potential accident-management strategies. Among the sensitivity studies of interest was one that considered whether a safety-relief valve (SRV) might stick open during the course of a severe accident. The SOARCA analyses concluded that it would be likely that an SRV would seize while the core melt was in progress, and the EPRI analyses largely adopted this assumption. While there is a technical basis for this assumption, it could, in some respects, constitute a non-conservatism. A stuck-open SRV would direct aerosols generated during core degradation to the suppression pool, where they would be scrubbed. This would tend to lower the magnitudes of releases from later drywell failures. To explore the impact of this treatment, the EPRI analyses included a sensitivity case that assumed none of the SRVs seized. For alternatives for which this change would have an impact, the result was approximately a 20% increase in the risk of latent-cancer fatality (well within the uncertainty range associated with severe-accident progression).

Another sensitivity study investigated the timing of drywell shell failure. As noted above, for the base-case, EPRI analyses indicate there would be essentially no water present in the drywell at the time of failure of the RPV. The MAAP code treats the molten core material as a liquid that would flow relatively quickly across the floor of the drywell. When the molten core material reached the steel drywell shell, it could lead to melting through of the shell. Melt-through of the drywell shell was calculated to occur within 15 min of vessel failure. Similarly, the SOARCA analyses assumed that shell failure occurred very shortly after vessel breach.

In the sensitivity case, the time to melt-through of the drywell shell was increased from 15 min to 10 hr. This long delay was introduced to maximize the potential benefit of an engineered filter. The result was a modest reduction in the amount of cesium released to the atmosphere. If the drywell shell eventually failed, it would still result in a significant release. While some of the fission products would have been deposited on cooler surfaces, they would tend to re-vaporize over time after the containment failure.

Another sensitivity that relates to the SOARCA analyses addressed variations in population sizes near the BWRs with Mark I containments. It was determined that the reference plant was near the upper range of such BWRs (with the second largest population). The sensitivity study explored the impact of a 30% increase in the population within 50 miles (80.5 kilometers) of the plant. The result of the sensitivity analysis was only a 1% increase in the risk of latent-cancer fatality within the 10-mile radius associated with the QHO.

A fourth sensitivity study of interest assessed the effectiveness of evacuation. In the baseline cases, for all of the alternatives, it was assumed that 99.5% of the population within 10 miles (16.1 kilometers) of the site would be evacuated (consistent with the assumption made in the SOARCA studies). The sensitivity study examined evacuation rates of 100%, 99.5%, 95%, and 0%. The results are illustrated in Figure 9 for the base case and three alternatives:

- *Alternative 2A* – provision for the addition of water to the RPV during a severe accident but without addition of a drywell vent.
- *Alternative 3A* – severe-water addition to the drywell instead of the RPV, but also without addition of a drywell vent.
- *Alternative 5B* – severe-water addition to the drywell and addition of a large engineered containment filter.

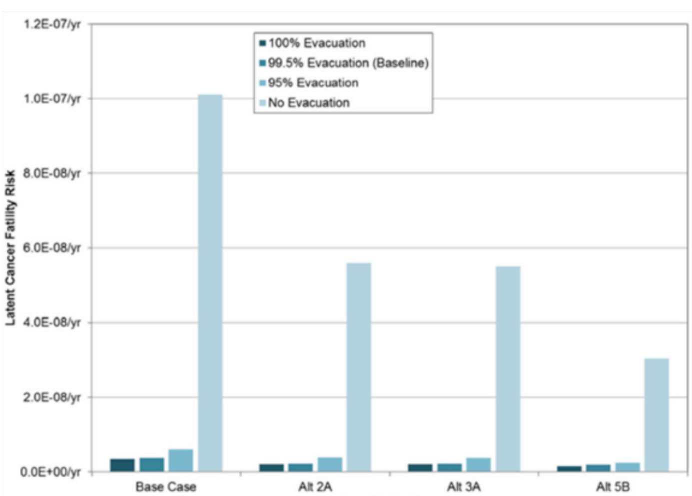

Figure 9 – Sensitivity of Risk of Latent Cancer Fatality to Effectiveness of Evacuation (Figure 4-15 from EPRI 3002003301 [21])

As can be seen, modest changes in the effectiveness of evacuation do not have much impact on the risk of latent-cancer fatality. If, however, there were no evacuation, the risk would be as much as 30 times higher for the base case. For the cases in which water can be added to the drywell, the risk is reduced. As would be expected, the risk impact assuming no evacuation would be smallest for the case with an engineered containment filter. But even in that extreme case, the difference relative to the two cases with water addition to the drywell is only a factor of two; the releases would still be most important for the cases in which the containment failed such that the suppression pool (for alternatives 2A and 3A) or the engineered filter (alternative 5B) would be bypassed, and therefore not effective in scrubbing releases.

In summary, the conclusions from the CPRR studies (both by NRC and EPRI) were largely consistent with those from SOARCA. All of the analyses found that the risk of latent-cancer fatalities was orders of magnitude below the QHO. The results of a variety of sensitivity studies exploring uncertain aspects of accident response provided no indication of significant deviation from this finding.

The CPRR analyses went further in investigating possible enhancements to the ability to manage a severe accident such as one caused by an ELAP. The analyses determined that being able to add water to reduce the potential for failure of the drywell for a BWR with a Mark I containment, and the resulting ability to scrub releases from either the wetwell or the drywell, would constitute more effective measures. These provisions would also be necessary for an engineered filter system to be effective. But if they were put in place, they would accomplish nearly the full objective of an engineered filter, such that the additional benefit from an engineered filter would be marginal.

Relevance of the Results to Different Plants

The analyses performed by NRC for SOARCA and by NRC and EPRI following the Fukushima accidents were based on specific plants, and the consequence calculations reflect the characteristics for their respective sites. A natural question arises regarding the applicability of the results and conclusions to other plants and sites.

One dimension of the variability among plants at different sites is the relative contributions of core-damage sequences. Both the SOARCA studies and the post-Fukushima CPRR analyses considered only a portion of the core-damage scenarios that would comprise the full risk profile. In both cases, the scenarios selected represented

Table 3 – Comparisons of Risks to QHOs for Assumed Frequencies

Scenario	Assumed Accident Frequency	Conditional Risk1	Computed Risk	Quantitative Health Objective	Margin to QHO
Individual risk of early fatality	1×10^{-5} (LERF)	3×10^{-7}	3×10^{-12}	5×10^{-7}	> 100,000
Individual risk of latent-cancer fatality (using risk of worst-case scenario)	1×10^{-4} (CDF)	3×10^{-4}	3×10^{-8}	2×10^{-6}	~ 70
Individual risk of latent-cancer fatality (using frequency-weighted average risk)	1×10^{-4} (CDF)	5×10^{-5}	5×10^{-9}	2×10^{-6}	> 300

Note
1. Conditional risk from SOARCA study for Surry [12]

those likely to present the more severe challenges to the containment, and consequently to lead to the largest offsite consequences. In that sense, the implications can be extended to encompass a broader set of accidents that might be more likely but would generally produce lower consequences.

Another way to view this impact is to assume that the core-damage frequency and the frequency of large, early release are at approximately the levels of their corresponding subsidiary objectives (1×10^{-4}/year for CDF and 1×10^{-5}/year for LERF). Recall that these measures were established as surrogates for the risk of latent-cancer fatalities and early fatalities, respectively (primarily to simplify analyses and to avoid the need for a full Level 3 PRA). Combining the conditional risk values from SOARCA with these assumed frequencies yields the results summarized in Table 3.

As can be seen from these results (and as noted earlier in this paper), the SOARCA studies computed very low risks of early fatality. The results, even assuming that the consequences applied for a full range of early releases at a frequency of 1×10^{-5}/year, indicate a large margin to the quantitative health objective.

The results for risk of latent-cancer fatalities assuming that all core-damage scenarios produce consequences comparable to those in the SOARCA studies (at a frequency of 1×10^{-4}/year) also indicate a large margin to the quantitative health objective. Note that two sets of results are presented for perspective. The first set reflects the use of the consequences for the worst-case scenario in SOARCA. The second set employs the risk weighted by the relative frequencies of all of the scenarios considered in SOARCA. Note, however, that even in the frequency-weighted case, the accidents considered were selected from among the more important severe accident scenarios for the specific plants considered, and do not necessarily represent more frequent, less challenging accidents that would be captured in a full PRA.

Another area in which variability among plants would be of potential interest would be with respect to the population density and other attributes of plant sites that could affect risk (e.g., local meteorology, land-use patterns, etc.). The three plants used in the SOARCA analyses (Peach Bottom, Surry and Sequoyah) are in areas that are not at the upper end in terms of the size of the nearby population. Because the quantitative health objectives are stated in terms of individual risks, population density does not necessarily have a direct impact on the risk measures. Population density does, however, affect evacuation speeds, so it is a factor for PRA Level 3 consequences. Likely of more importance is the characteristics of the weather patterns in the vicinity of the plant, especially in relation to the locations of population centers. That is, if there is a significant chance that the winds will be in the direction of populations or that precipitation will cause increased deposition in particular locations, the risk can be higher irrespective of the total population near the plant.

Population density was investigated in sensitivity studies in both the NRC and EPRI analyses, as noted in the discussions above. In both cases,[2] the risk of latent-cancer fatality was determined to be insensitive to population density.

To investigate this aspect further, two additional sensitivity studies were performed. The first sensitivity study coupled the release characteristics for the representative plant in the EPRI study with the site characteristics for the BWR with the highest nearby population. The results were

- An increase of approximately 50% in the mean risk of early fatality.
- Slightly less than doubling of the mean risk of latent-cancer fatality.

Even for such increases, the margins to the QHOs remain large.

2 Refer to Section 4.4.3 of the NRC CPRR study [20] and Section 4.1.4 of the EPRI study [21] for further details related to the sensitivity to population density.

A second, more extensive sensitivity study was performed to extend the consideration of site variability for this white paper by calculating the conditional risk (that is, conditional on the occurrence of a range of releases) for plants across the U.S. This sensitivity study made use of a set of consequence calculations performed in 2005 for the Risk Analysis and Management for Critical Asset Protection (RAMCAP) program [22]. As part of that program, offsite consequences were assessed for a range of release categories. The original calculations, however, did not explicitly address the QHOs' specific conditions (i.e., individual early fatalities within 1 mile of the site and individual latent cancer fatalities within 10 miles, i.e., 16.1 kilometers). Therefore, the original results were used to calculate these two specific risk measures for each of the sites in the U.S.

These calculations were performed using an older characterization of release magnitudes and other inputs that are not reflective of the consequence calculation advances made in the SOARCA analyses. Nevertheless, they do present one picture of the variability in risk by site. For this perspective, the conditional consequences for individual latent cancer fatalities across all releases were normalized to two reference plants, and the populations within 10 miles (16.1 kilometers) were similarly normalized to those for the two plants. The result for this sensitivity study is one depiction of the potential range in consequences as a function of population. The results for the two references plants are shown in Figures 10 (for Peach Bottom) and Figure 11 (for Sequoyah).

The red circled point in Figure 10 corresponds to the result for the Peach Bottom plant for which comparisons are made (Peach Bottom was a representative plant in both the SOARCA and EPRI CPRR studies). The largest conditional risk is about a factor of 6 higher than that for Peach Bottom, and it is for a site with the 11th largest population within 10 miles (16.1 kilometers). The risk for the plant with the highest population is approximately a factor of 5 higher than that for Peach Bottom.

Similarly, the red circled point in Figure 11 represents the anchor point for the Sequoyah plant. Sequoyah has nearly twice the population within 10 miles (16.1 kilometers) of that for Peach Bottom. The plant with the highest risk is approximately twice that for Sequoyah, and the plant with the highest population is less than a factor of two higher in latent-cancer risk than Sequoyah.

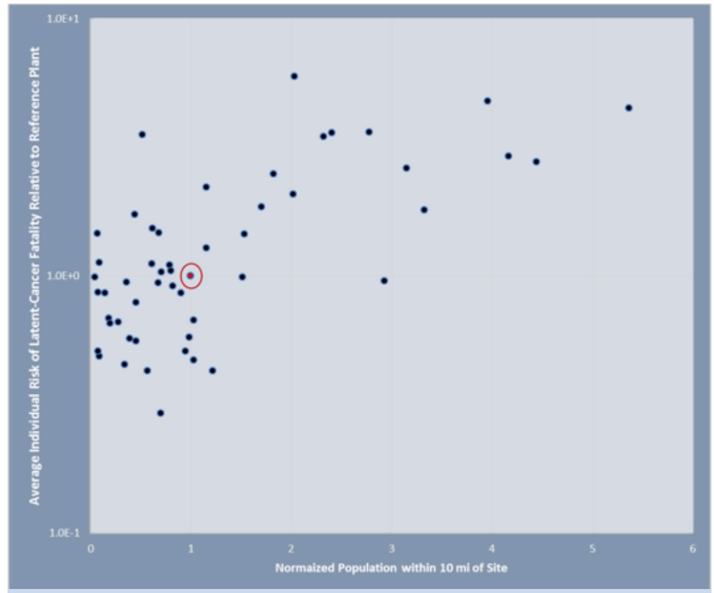

Figure 10 – Variability of Conditional Risk of Latent-Cancer Fatalities for Sites in the U.S. Relative to Peach Bottom

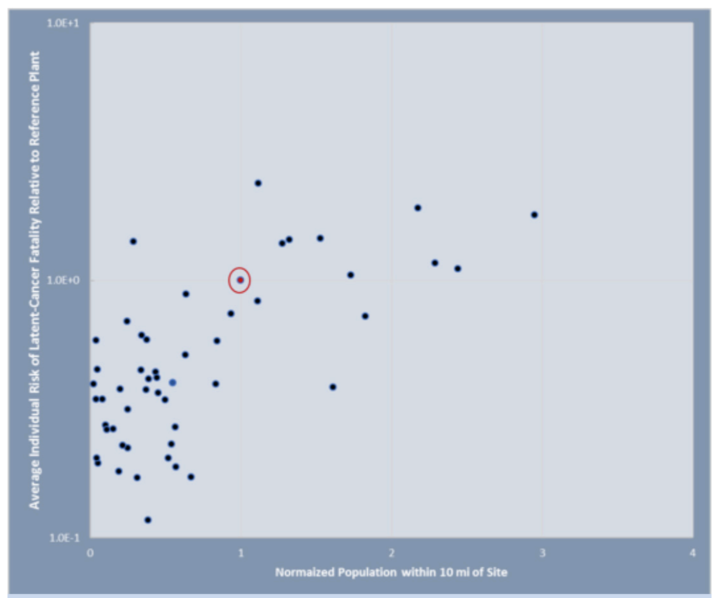

Figure 11 – Variability of Conditional Risk of Latent-Cancer Fatalities Relative for Sites in the U.S. Relative to Sequoyah

While more extensive plant- and site-specific calculations of risk could be made, these sensitivity studies would indicate that there would still be very large margins to the qualitative health objectives for any plant operating in the U.S.

Summary and Conclusions

The most recent information available indicates that there are significant margins between the quantitative representations of the U.S. NRC's safety goals (that is, the quantitative health objectives – QHOs) and the subsidiary objectives widely used in considering risk-informed applications in the U.S. (that is, CDF and LERF). The analysis also indicates that the margins between the levels of risks posed by specific plants and the QHOs are large as well.

For example, consider the margins with respect to the risk of early fatality. As Table 3 illustrates, if a plant were to have a frequency of large early release at the subsidiary objective of 1 x 10^{-5}/year, with consequences corresponding to the most serious releases that might be anticipated, the margin to the QHO is a factor on the order of 100,000 or more. This is primarily because releases large enough to produce early health effects and rapid enough to occur before the population adjacent to the site has been evacuated are extremely unlikely (as concluded by SOARCA). One implication of these findings is that LERF may not be a very meaningful risk metric. A plant could have a substantially higher frequency of large releases but still not pose a risk of early fatalities to individuals within 1 mile of the site boundary.

Table 3 also provides a perspective on the margins available with respect to the risk of latent-cancer fatality. The frequency of core damage is assumed to be at the subsidiary objective of 1 x 10^{-4}/year, and the consequences were calculated for releases generally associated with the more important severe accident scenarios (station blackout, interfacing-systems LOCAs, etc.). In this case, the margin to the QHO is on the order of a factor of 300 (and about 70, if the entire core-damage profile resulted in a release comparable to the more severe accidents evaluated by SOARCA).

Additional perspective on the risk of latent-cancer fatality is available from Figure 12, which was published as Figure ES-3 in NUREG-1935 [12]. This figure compares the risks of fatality due to latent cancer to the risks that would be obtained using an older set of source terms[3] and, more significantly, to the NRC's QHO. The graph indicates that, for the frequency of particularly challenging accidents for Surry and Peach Bottom, the margin to the QHO is more than a factor of 1000.

Figure 12 illustrates the large impact of the changes in understanding of severe accidents and their potential to lead to offsite health effects that have taken place through nearly four decades of research and analysis.

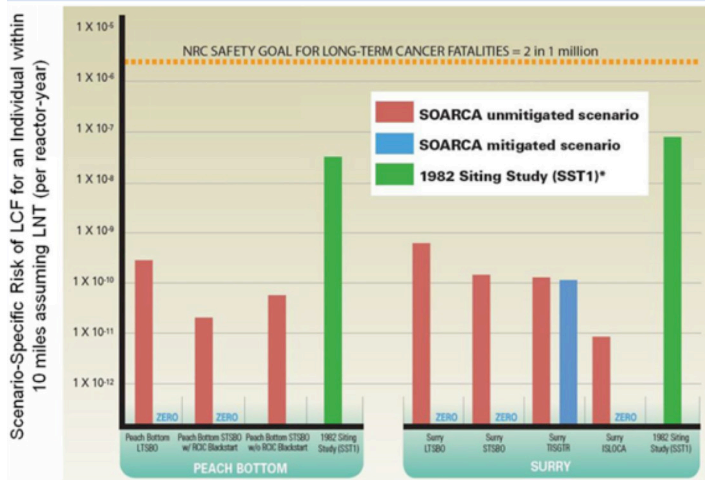

Figure 12 – Perspective on Risks of Latent Cancer Fatality (Figure ES-3 from NUREG-1935 [12])

This evolving understanding related to severe accidents is summarized in Figure 13. The initial bounding assessments to confirm the adequacy of the surrogate or subsidiary QHOs estimated margins on the order of 2 to 5. Using the updated understanding described in the SOARCA analyses and other insights into severe accidents, these margins now appear to be on the order of a factor of 100 for individual risk of latent-cancer fatality and more than 10,000 for individual risk of early fatality. The technical insights indicate that large margins appear to exist with respect to the risk of early fatality (i.e., relative to the subsidiary objective for large early release frequency).

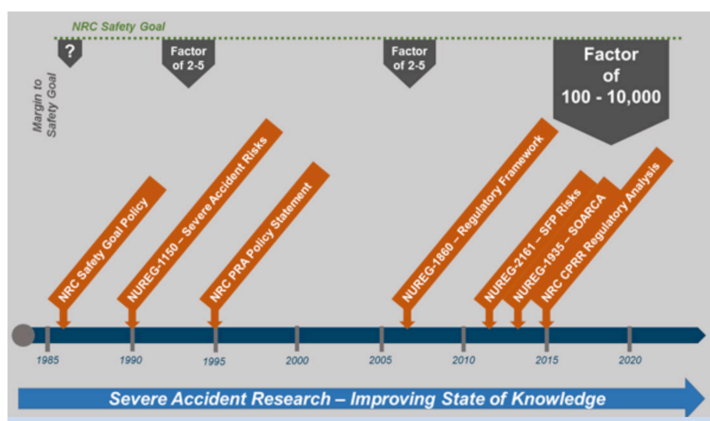

Figure 13 – Impact of Evolving Understanding of Severe Accidents on Perceived Risk Margins

3 When the 1979 accident at Three Mile Island progressed beyond the design basis with respect to the extent of core damage, but resulted in releases from the fuel and from containment that were far smaller than assumed in the design basis, the NRC re-assessed accident source terms. The result was the 1982 Siting Study (SST1) [23]. While the revised source terms were smaller than previous design-basis assumptions, they are still far larger than those assessed in SOARCA or in NUREG-1150 and in PRAs performed since the mid-1980s.

The purpose of this white paper is to contribute to an improved understanding of risk-informed decision-making. The safety goals and subsidiary objectives were established with the intent that they would be applied relative to mean estimates of risk. Several of the assessments presented are bounding in the sense that the frequencies were assumed to be at the subsidiary objectives (LERF of 1 x 10^{-5}/year; CDF of 1 x 10^{-4}/year). If the accident frequencies for a particular plant were calculated to be at these values, the vast majority of the accident scenarios that contributed to those frequencies would be less severe than those represented by the releases assumed in calculating the consequences in Table 3. This would suggest that the margins presented in this white paper may be underestimated for plants in general.

Further, the margins provide an opportunity to consider the role of uncertainty in decision-making that takes into account risk information. The need to consider uncertainties when relying on PRA methods has been discussed extensively, dating back to the issuance of the NRC's Policy Statement on the Use of PRA Methods in Nuclear Regulatory Activities [24] in 1995. In this Policy Statement, the NRC stated the position to increase the use of PRA in regulatory matters to include, among other aspects, uncertainty analyses within the bounds of the state-of-the-art, to reduce unnecessary conservatisms in regulatory practices. In addition, this document states that the use of the safety goals for nuclear power plants and subsidiary numerical objectives need to include an appropriate consideration of uncertainties in making regulatory judgments.

More recent guidance on the treatment of uncertainties in PRA include the update to NUREG-1855 [25] and companion EPRI documents [26, 27] on this topic. It is also addressed further in the context of decision-making when it is necessary to aggregate various risk contributors [28].

Many of these references and, particularly that provided in NUREG-1855, are concerned with the need to understand uncertainties when the quantitative risk results might have the potential to lead to one or more of the subsidiary objectives being exceeded. From the information presented here, the insights are that only in the most extreme cases could uncertainties play a significant enough role in this respect, at least with regard to cases in which the CDF approaches (or even exceeds) the subsidiary objective.

Other lower-tier objectives and guidelines are employed in a variety of risk applications. For example, the guidance presented in Regulatory Guide 1.174 [10] is typically applied in assessing the acceptability of formal risk-informed changes to the licensing basis for operating nuclear power plants. Using this guidance, changes that result in small increases in risk may be found to be acceptable. "Small" in this case typically refers to a change in core-damage frequency of 10^{-6}/year or less, or an increase in LERF of 10^{-7}/year or less (essentially 1% of the levels of the respective subsidiary objectives). Given the large margin between the subsidiary objectives and the QHOs, these changes can be seen to be extremely small with respect to the impact on adequate safety.

In conclusion, as risk applications continue to be employed and, perhaps, expanded, it is important that the overall perspectives on risk, such as those presented in this paper, be kept in mind to help ensure decisions are made in an objective and effective manner.

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