METHODS AND STRATEGIES FOR FUTURE REACTOR SAFETY GOALS

Dissertation

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ABSTRACT

There have been significant discussions over the past few years by the United States Nuclear Regulatory Commission (NRC), the Advisory Committee on Reactor Safeguards (ACRS), and others as to the adequacy of the NRC safety goals for use with the next generation of nuclear power reactors to be built in the United States. The NRC, in its safety goals policy statement, has provided general qualitative safety goals and basic quantitative health objectives (QHOs) for nuclear reactors in the United States. Risk metrics such as core damage frequency (CDF) and large early release frequency (LERF) have been used as surrogates for the QHOs.

In its review of the new plant licensing policy the ACRS has looked at the safety goals, as has the NRC. A number of issues have been raised including what the Commission had in mind when it drafted the safety goals and QHOs, how risk from multiple reactors at a site should be combined for evaluation, how the combination of a new and old reactor at the same site should be evaluated, what the criteria for evaluating new reactors should be, and whether new reactors should be required to be safer than current generation reactors.

As part of the development and application of the NRC safety goal policy statement the Commissioners laid out the expectations for the safety of a nuclear power plant but did not address the risk associated with current multi-unit sites, potential

modular reactor sites, and hybrid sites that could contain current generation reactors, new passive reactors, and/or modular reactors. The NRC safety goals and the QHOs refer to a "nuclear power plant," but do not discuss whether a "plant" refers to only a single unit or all of the units on a site. There has been much discussion on this issue recently due to the development of modular reactors. Additionally, the risk of multiple reactor accidents on the same site has been largely ignored in the probabilistic risk assessments (PRAs) done to date, and in most risk-informed analyses and discussions.

This dissertation examines potential approaches to updating the safety goals that include the establishment of new quantitative safety goal associated with the comparative risk of generating electricity by viable competing technologies and modifications of the goals to account for multi-plant reactor sites, and issues associated with the use of safety goals in both initial licensing and operational decision making. This research develops a new quantitative health objective that uses a comparable benefit risk metric based on the life-cycle risk of the construction, operation and decommissioning of a comparable non-nuclear electric generation facility, as well as the risks associated with mining and transportation. This dissertation also evaluates the effects of using various methods for aggregating site risk as a safety metric, as opposed to using single plant safety goals. Additionally, a number of important assumptions inherent in the current safety goals, including the effect of other potential negative societal effects such as the generation of greenhouse gases (e.g., carbon dioxide) have on the risk of electric power production and their effects on the setting of safety goals, is explored. Finally, the role risk perception should play in establishing safety goals has been explored. To complete this evaluation, a new method to analytically compare alternative technologies of generating electricity was developed, including development of a new way to evaluate risk perception, and a new method was developed for evaluating the risk at multiple units on a single site.

To test these modifications to the safety goals a number of possible reactor designs and configurations were evaluated using these new proposed safety goals to determine the goals' usefulness and utility. The results of the analysis showed that the modifications provide measures that more closely evaluate the potential risk to the public from the operation of nuclear power plants than the current safety goals, while still providing a straight-forward process for assessment of reactor design and operation.

DEDICATION

To my family, friends, teachers and colleagues who have seen me through the many hard times and always had faith in me (Jacqueline, Harold, Craig, Dorothy, Vicki, Bruce, Denny, Ed, Pat, Dave, John, Don, Peter, Madeline, Rosemary, Hash, Richard, Bob, Don, Tunc, Audeen, Brian, and Rich).

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PUBLICATIONS

J. Kirschenbaum, P. Bucci, M. Stovsky, D. Mandelli, T. Aldemir, and S. A. Arndt, "A Benchmark System for Comparing Reliability Modeling Approaches for Digital Instrumentation and Control Systems," *Nuclear Technology,* Vol. 165, No. 1, pp 53-95, January 2009

T. Aldemir, D.W., Miller, M. Stovsky, J. Kirschenbaum, P. Bucci, L.A. Mangan, A. Fentiman and S.A. Arndt, "Methodologies for the Probabilistic Risk Assessment of Digital Reactor Protection and Control Systems," *Nuclear Technology,* Vol. 159, No. 2, pp 167-191, August, 2007

C.S. Smidts and S.A. Arndt, "Nuclear Power Plant Control Systems", *Encyclopedia of Computer Science and Engineering*, edited by Benjamin Wah, Wiley, New York, New York, 2006

S.A. Arndt, "Simulation and Simulators - Their Role in Science and Society", *Encyclopedia of Microcomputers*, Marcel Dekker, Inc., New York, New York, 1994

T. Aldemir, S. A. Arndt and D. W. Miller, "Simulation of the Transient Response of Ionization Chambers to Bias Voltage Perturbations", *Nuclear Technology*, Vol. 76, No. 2, February, 1987

FIELDS OF STUDY

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CHAPTER 1: INTRODUCTION

The Nuclear Regulatory Commission's (NRC's) safety goals are described in the NRC Safety Goal Policy Statement released in August 1986 [1]. The development of the policy statement began not long after the Three Mile Island accident, and was based on research on quantitative risk assessment and societal risk acceptance that had been done to that point in time. The Safety Goal Policy Statement was the NRC's first attempt to explicitly come to grips with the need to develop quantitative assessment of risk as part of a regulatory structure. Traditionally, the NRC, like most regulatory agencies, has developed deterministic regulatory rules and guidelines based on best engineering practices and judgment. These methods provided guidance as to what engineering analysis needed to be completed to demonstrate adequate protection of the health and safety of the public, and provided appropriate acceptance criteria for the analysis. Although this approach is a common and effective method for regulating activities, it does not effectively deal with the natural tendency of stakeholders to wish for ever increasing levels of safety. After the Three Mile Island accident, the NRC added a significant number of additional regulatory requirements. The degree of additional safety benefit provided by these requirements has been long debated.

With the development, in the mid-1970s, of probabilistic safety assessment tools and the completion of the first study using probabilistic techniques to estimate the frequency of accidents and their consequences, the Reactor Safety Study, also known as WASH-1400 [2], the NRC had a tool that could be used to directly estimate the effects of nuclear power operation on the safety of the public. The primary propose for the NRC in developing qualitative and quantitative safety goals was to use these techniques and tools to help articulate a level of acceptable risk, in other words, to define "how safe is safe enough."

While the safety goals provided a metric to address the question of “how safe is safe enough,” practical implementation of the NRC’s guidance has taken many years to implement and has proved to be difficult. As a result, the NRC developed a number of other metrics to use as surrogates for the safety goals to use in regulatory decision-making. Over the past twenty years, the NRC has developed a number of practical uses for the safety goals, and has interpreted their meaning to support regulatory decision making. However, issues associated with the safety goals have continued to be raised by the NRC’s external scientific advisory committee, the Advisory Committee on Reactor Safeguards (ACRS), the nuclear industry and others as to how well the safety goals meet the current and future needs of the nuclear industry and whether they should be revised. This discussion is most frequently directed to the adequacy of the safety goals for use with the next generation of nuclear power reactors to be built in the United States. This dissertation examines potential approaches to update the safety goals, including the establishment of a new quantitative goal associated with the comparative risk of generating electricity by viable competing technologies and will perform some preliminary case studies to examine the need and effectiveness of such a quantitative goal.

1.1. Problem Description

The nuclear industry is currently struggling with how to interpret the safety goal policy statement [3-4] in light of the advanced reactor policy statement [5] and the current regulatory review of advanced light water reactors, such as the Westinghouse Advanced Passive 1000 (AP1000), the General Electric Economic Simplified Boiling Water Reactor (ESBWR) and the AREVA U.S. Evolutionary Power Reactor (EPR), as well as next generation nuclear reactors such as high temperature gas reactors. The advanced reactor policy statement indicates the NRC expectation that the new generation of nuclear power plants to be built in the United States should be safer than the generation of nuclear power plants operating today. However, the policy statement provides no quantitative goals or metrics to evaluate this expectation. Additionally, if two plants, one new and one of the current generation, are to be placed on the same site, the question of how these plants are to be evaluated becomes still more challenging. The safety goals are written as goals for the safety of the public. If two plants are on the same site, the population that might be impacted is the same. Should these co-located power plants be evaluated differently just because one is newer than the other? In their letter to the NRC Commission of September 21, 2005, the ACRS [4] posed these and other issues in the form of nine questions.

1) What is the appropriate type of safety goal?

2) Should the safety goals be used primarily as design goals or as a plant (site) specific measure of operational safety?

3) Should the safety goals limit the frequency of accidents as well as the consequences of the accident?

4) What are the appropriate measures of safety (Core Damage Frequency (CDF), Large Early Release Frequency (LERF), Large Release Frequency (LRF), etc.)?

5) What should the acceptance criteria be?

6) How should risk from multiple reactors at a site be combined for evaluation by suitable criteria?

7) How should the combination of new and old reactors at a site be evaluated by these criteria?

8) How should compliance with these criteria be demonstrated?

9) How do you evaluate the uncertainties associated with the analysis?

A safety goal strategy needs to be developed to help support the construction, operation and regulation of the next generation of nuclear power plants in the United States. This strategy must answer the questions of how to include both the current generation of nuclear power plants and future power plants, how to account for the safety at sites with multiple power plants, and how to include other considerations such as power level and the comparative risks of other methods of generating electricity.

1.2. Objectives and Scope

Alvin Weinberg termed issues, such as the setting of qualitative and quantitative safety goals, “trans-scientific” in that they are beyond the realm of fundamental science and engineering since they cannot be solved by science and engineering alone but also need societal policy to answer them [6]. From the time the NRC first articulated its safety goal policy statement, it has been challenged to relate the high-level qualitative goals to goals that are more easily measured. To resolve this issue the NRC developed the CDF and LERF subsidiary goals. However, it is not clear that these subsidiary goals are applicable to advanced non-light water reactors designs being proposed for future development and deployment in the United States. As discussed above, the NRC’s policy statement on the safety of advanced reactors articulates an expectation that the next generation of reactors built in the United States should be safer than the current reactor fleet. This dissertation examines the existing probabilistic safety goals and the subsidiary goals to determine their applicability to future designs of individual plants and to mixes of multiple plant designs on a single site. A new quantitative safety goal associated with comparative societal risks of alternative modes of electricity generation is also developed and examined.

The dissertation's research consists of two efforts. The first effort is to develop a new safety goal that will address the objective of ensuring that nuclear power imposes comparable, or less, risk to the public than risks posed by other, alternative methods of providing electric energy. The second effort is to modify the current safety goals to account for multi-reactors on a single site. The research uses the MELCOR Accident Consequence Code System (MACCS) to perform cases studies to examine the practicality and potential limitations of the new goal. Source terms for selected reactor types and site configurations are developed to support the consequence analysis. Where possible, available source term information is used; additionally, risk analysis is completed to support the evaluation of accident frequencies at multi-plant sites.

1.3. Overview

This dissertation comprises eight chapters. **Chapter 1** is a brief overview of the dissertation. In this chapter, the problem to be addressed is discussed and the scope and objectives of the dissertation, as well as the technical approach to the problem, are outlined.

Chapter 2 provides a brief background of the current safety goals, how they were developed, their limitations and the need for new and/or modified safety goals to address future reactors. The chapter also discusses the importance of the manner in which safety goals are interpreted with respect to multi-unit sites.

A review of the literature is presented in **Chapter 3**. The literature on the development and use of safety goals, as well as the previous research on development of safety goals and interpretation of safety goals, are discussed.

Chapter 4 discusses the way in which a new or modified safety goal might be developed and describes the development of the new comparative risk-benefit goal proposed in this research. Chapter 4 also discusses proposed modifications to, and a different interpretation of, the current safety goals in order to address the multi-unit issues.

Potential application issues related to the new and modified safety goals and their corresponding analyses are discussed in **Chapter 5**. Also discussed are the need to modify the subsidiary goals and the resulting implications to current regulatory guidance.

In **Chapter 6**, the proposed modified and new safety goals are demonstrated. Tests of the new goals include sample calculations reflecting the consequences of applying these modifications to several different reactor site configurations.

Chapter 7 provides a discussion of sensitivity studies and uncertainty analysis of the new goals and subsidiary goals and potential impacts on regulatory guidance to support use of the new goals.

In the final chapter, **Chapter 8**, conclusions are drawn from the analyses that have been performed. Recommendations as to future research are also provided.

CHAPTER 2: BACKGROUND

The NRC Safety Goal Policy Statement released in August 1986 [1] established three qualitative safety goals supported by two quantitative objectives. These supporting objectives are based on the principle that nuclear risk should not be a significant addition to other societal risks faced by the public of the United States. In discussing the safety goals, the Commission made clear that no deaths attributable to nuclear power plant operation are considered "acceptable." The Commission, in its Safety Goal Policy, stated the goals are acceptable risks, not acceptable deaths. The qualitative safety goals are:

> 1) 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,
> 2) Societal risk to life and health from nuclear power plant operation should not be a significant addition to other societal risk, and
> 3) Societal risk to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing alternative technologies.

The quantitative objectives (now usually referred to as the quantitative health objectives (QHOs)) were developed as a method of evaluating to what extent the qualitative safety goals had been achieved. The quantitative objectives are:

> 1) The risk to an average individual in the vicinity of a nuclear power plant of prompt fatality that might result from reactor accidents should not exceed one-tenth of one percent of the sum of prompt fatality risk resulting from other accidents to which members of the U.S. population are generally exposed, and
> 2) 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 of the sum of cancer fatality risk resulting from all other causes.

The QHOs are stated in terms of public health risk, with one QHO addressing individual risk and the other addressing societal risk[1]. The risk to an individual is based on the potential for death resulting directly from a reactor accident, i.e. a prompt fatality. The societal risk is stated in terms of nuclear power plant operations, as opposed to accidents alone, and addresses the long-term impact on those living near a nuclear plant. In both cases, the Commission based its acceptable level of risk on a comparison with other types of risk encountered by individuals and by society from other causes. The goals were expressed in qualitative terms, perhaps so the goals could be more easily understood by the general public. The Commission chose the concept that the relative

[1] Although the second QHO is typically referred to as a societal goal, it is measured in terms of the risk to an individual. It is arguable as to whether this is truly a societal goal or just a second individual risk goal.

consequences of nuclear power plant operation should not result in significant additional risks to life and health. In the QHOs they chose the acceptance criteria of "less than one tenth of one percent" as being equivalent to the "no significant additional risk" criteria provided in the qualitative goals.

Although the safety goals have been used to define the concept of "how safe is safe enough", it's important to note that the safety goals and the QHOs themselves have never been directly reflected in the NRC's regulations. The goals were originally developed to provide guidance as to the level of "public protection which nuclear plant designers and operators should strive to achieve [7]." They were also meant to provide guidance to the NRC staff[2] for use in the regulatory decision-making process. However, the Commission was clear that the safety goals were not meant "to serve as a sole basis for licensing decisions [8]." In fact, at the time, the Commission disclaimed any intent to use the goals in making plant-specific regulatory decisions.

2.1 History of Nuclear Power Plant Safety Goals

The development of the NRC's safety goals dates from the recommendations of the President's Commission after the accident at Three Mile Island [9]. In the response to the recommendations in that report, the NRC stated that it was "prepared to move forward with an explicit policy statement on safety-cost tradeoffs in the NRC safety

[2]The NRC is comprised of the Commissioners, a group of one to five presidential appointments that provides the policy level guidance for the NRC staff, which does the day to day work of the NRC. When referring to the NRC it is sometime necessary to distinguish between the Commissioners and the staff, because the staff might recommend a particular course of action on a policy matter to the Commissioners that they might not choose to implement. In this research the entire NRC will be referred to as the "NRC," the NRC Commissioners will be referred to as the "Commission," the "Commissioners" or the "NRC Commissioners," and the NRC staff will be referred to as the "NRC Staff."

decisions [1].” The Advisory Committee on Reactor Safeguards (ACRS) in a letter dated May 16, 1979, also recommended that consideration be given to establishment of quantitative safety goals for nuclear power plants. In October, 1980, the ACRS provided the Commission with a preliminary proposal, NUREG-0739 "An Approach to Quantitative Safety Goals for Nuclear Power Plants," [10] for a possible approach to quantitative safety goals. In that proposal the safety criteria included limits on the frequency of occurrence of certain hazardous conditions within the reactor, limits on risks to individuals of early death or delayed death due to cancers arising from a reactor accident, limits on overall societal risk of early or delayed death, and application of the “as low as reasonably achievable” approach combined with a cost-benefit criterion for preventing premature death. In addition to this proposal, the NRC solicited and reviewed information and recommendations provided by workshop discussions at NRC-sponsored workshops held in Palo Alto, California and Harpers Ferry, West Virginia, in 1981. The workshops addressed general issues involved with developing safety goals and specific input based on a discussion paper which presented proposed safety goals. The NRC staff provided the Commission, for its consideration, a discussion paper on safety goals in November 1981 and a revised safety goal report in July 1982 [1]. The Commission also took into consideration input it received from the public in response to a proposed Policy Statement and comments received in connection with a 2 year evaluation period for a revised Policy Statement issued in March 1983. Based on the information from the evaluation of the comments and additional input from stake holders including the ACRS, the Commission published the final Safety Goal Policy Statement [1] in 1986.

In 1990, the Commission provided additional guidance to the NRC staff regarding the safety goals, endorsing the surrogate measures concerning the frequency of core damage accidents and large early releases of radioactivity. The numerical value of one-in-ten-thousand years (10^{-4}) for CDF was chosen as a "very useful subsidiary benchmark [8]." In addition, a conditional containment failure probability (CCFP) of one-tenth (0.1) was approved for application to evolutionary (advanced) light water reactor designs. This resulted in a LERF of one in one-hundred-thousand years (10^{-5}), because for most accident sequences, containment failure is necessary for a large release. As can be seen, the 10^{-5} acceptance criterion for LERF is derived from the product of CDF and CCFP. Although on a per plant basis this is a very unlikely event, for a fleet of 100 plants we would expect a large early release once every thousand years ($10^{-5} \times 10^{2} = 10^{-3}$). These values ($10^{-4}$ for CDF and 0.1 for CCFP) were tested to see if they adequately represented the QHOs for light water reactors and were shown to limit the offsite consequences to below the QHOs [11]. These values have evolved into the subsidiary safety goals of 10^{-4} for CDF and 10^{-5} for LERF, that are used in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Bases [12]," and which form the basis for risk-informed regulatory decision-making.

In its staff requirements memorandum related to safety goals, the Commission provided instructions to the NRC staff as to how it wanted the safety goals to be implemented [8]. The Commission indicated that "the safety goals should be applied to all designs; independent of the size of the containment or character of a particular design approach to the release mitigation function." It also indicated that the use of subsidiary objectives for CDF and CCFP were an appropriate way of measuring and managing a plants adherence to the safety goals. Additionally it indicated that "the subsidiary objectives should anchor, or provide guidance on 'minimum' acceptance criteria for prevention …and …mitigation …and thus assure an appropriate multi-barrier defense-in-depth balance in design[3]." The Commission did not want the subsidiary objectives to become de-facto requirements or to be considered replacements for the deterministic regulations. One example of this was the specific guidance on not removing requirements for reactor containments based on the safety goals. The Commission did not want the low risk of core damage to be used as a justification for reducing the requirements on reactor containment, even when a sufficiently small value of CDF implied that the LERF was within the safety goals. They also instructed the NRC staff to use the safety goals as a way to define "how safe is safe enough" and indicated that the safety goals should "be seen as guidance on how far to go when proposing safety enhancements, including those to be considered under the Backfit Rule[4].

In 1996 the ACRS issued a letter to the Commission [13] about risk-informed, performance-based regulation and related matters which recommended, among other things, that the safety goals be used to derive guidelines for plant-specific actions, that the subsidiary objectives (goals) for CDF (10^{-4} per reactor-year) should be stated as a

[3] The NRC has traditionally used both the concept of prevention and mitigation to ensure that the likelihood of off-site consequences due to a reactor accident is as small as possible. This has included regulations to reduce the frequency of accidents, strengthen the capability of safety systems to mitigate accidents, containment systems to prevent the release of radiation, and siteing criteria to reduce the effected off-site population.

[4] The Backfit Rule, 10CFR50.109, requires that new requirements represent a substantial increase in protection to public health and safety or common defense and security whose costs are justified in light of this increased protection. A backfit analysis uses traditional cost/benefit analysis to determine if the cost of the backfit is justified and uses an evaluation against the safety goals to determine when a regulatory requirement should not be imposed generically on nuclear power plants because the residual risk is already acceptably low.

fundamental safety goal and that the staff should consider the appropriate treatment of temporary changes in risks caused by configuration changes.

As a result of this letter and other on-going work aimed at implementing risk informed regulation, the NRC in 2000 reviewed the safety goals [14] to determine if modifications were needed, and recommended to the Commission that several changes be made to the safety goals. These changes included plant-specific usage of the safety goals, elevating the prevention of severe core damage accidents from a quantitative to a qualitative goal, incorporation of additional guidance on uncertainty, addition of a statement regarding the role of defense-in-depth in a risk-informed regulatory framework, and the incorporation of the LERF subsidiary goal of 10^{-5} per reactor-year. The recommendation to highlight (elevate) the prevention of severe core damage accidents by including them as an additional qualitative goal was included by the staff in their recommendations to highlight the widely held belief that even if there is little or no off-site safety consequences (no significant additional risk to the public) a severe core damage accident will have very detrimental effects on the industry and the public perception of nuclear power. In a staff requirements memorandum [15], the Commission approved all these recommendations except that of changing the goal of preventing severe core damage accidents to a qualitative goal, and requested the NRC staff to make the needed changes to the safety goals. However, when the NRC staff proposed the needed changes in 2001 [16], the Commission disapproved issuance of the revised safety goals [17].

The Commission again reviewed the safety goals as part of its development of a possible technology-neutral framework for new plant licensing. In this review the NRC staff also looked at the issues of multiple and modular new reactors on a single site and levels of safety between current and new reactors. The NRC staff recommended to the Commission [18] that the safety goals be implemented in such a way as to raise the QHOs for new plants to the status of minimum safety levels. They further recommended that the criteria for new plants be interpreted such that as a whole all of the new reactors (modular or multiple) at a site should comply with risk goals expressed in the QHOs. However, again in its staff requirements memorandum [19] on the subject, the Commission disapproved the NRC staff's recommendations[5].

2.2 Review of the Current Safety Goals and their Use

Over the last 24 years the application of the safety goals has become the foundation for many of the basic elements of the nuclear regulatory system in the United States. Although the Commission originally did not intend the safety goals to be directly used in regulatory decision making, over time the safety goals have evolved from a primarily philosophical abstract into very practical guidance used for the development of regulatory rules and guidance. Currently the safety goals serve as the basis for many regulations including, for example, the explicit consideration of risk in the NRC's Backfit Rule. The safety goals have also played an important role in NRC's development of risk-informed regulation. In Regulatory Guide 1.174 the safety goals and the associated subsidiary objectives are used to define the acceptance guidelines for risk informed regulatory decisions associated with changes to a plant's licensing basis. The

[5]The primary rational for not accepting this recommendation was the desire not to imply that the current plants were not already safe enough.

development of other regulatory guidance, including most of the risk-informed guidance and procedures used today in the nuclear industry, has also been affected by input based upon the safety goals. Accordingly, any changes to the safety goals need to be evaluated with respect to their possible effects on current regulatory practices.

In applying the QHOs a number of assumptions and common practices have developed. For the application of the QHO related to cancer fatalities, the NRC defined the population subject to significant risk as the population within 10 miles of the plant site. Although there have been a number of discussions, dating back to before the formal issuance of the safety goals, of possibly using 50 miles, the Commission has consistently chosen to maintain the 10 mile distance for this analysis. This choice effectively limits the effect that large populations exposed to very small doses has on the safety goals. Because in the quantitative assessment of this safety goal the total number of effects observed is divided by the total population within the radius of the zone, the smaller the radius of the zone evaluated the more stringent the criterion. In a true societal safety goal, treatment of the linear no threshold model can have a major effect on the perceived societal impact.

Additionally, the way the safety goals are applied in practice today is based upon the assumption that the goals are for a single plant. The subsidiary goals have been defined on a single reactor basis and with CDF and LERF as "per reactor year." Thus, the QHOs and the subsidiary goals do not depend on how many plants are on a site or on what kind of plants are on the site. As discussed above, in the past there have been some considerations related to modifying the safety goals to account for these issues, but they have never been included within the safety goals themselves or in the way in which the safety goals are implemented.

2.3 Need for New Safety Goals for Future Reactors

Although the safety goals have provided a very effective method of establishing a basis for evaluation of reactor safety and determining "how safe is safe enough", they possess some significant limitations and challenges, particularly with respect to future reactors. While the use of the subsidiary measures CDF and LERF have proven to be of value in implementing the Commission's safety philosophy, they tend to skew the focus of attention to severe reactor accidents. While it is true that PRA results indicate that the societal risk from nuclear power is dominated by accidents that have low frequencies and high consequences, the perception of risk on the part of the public is also influenced by events of low consequence (in terms of radioactive releases), but perceived to be of higher consequence, which occur at much higher frequencies. This is illustrated, for example, by the reaction following the steam generator tube failure at the Indian Point 2 station in February 2000 which was widely reported to have involved a release of radioactivity to the environment [20]. Although the release was determined to be so minor that the monitoring equipment around the plant could not detect it, nonetheless, there was an intense public reaction to the event. Safety goals that only focus on the worst case reactor accidents tend to not appropriately take into account the public's intense interest in all nuclear events. Without adequately addressing these concerns, these safety goals do not give the public an accurate view of the risk related to the use of nuclear power risk compared to other technologies that also generate electricity.

A safety goal strategy needs to support the construction, operation and regulation of the next generation of nuclear power plants in the United States and answers the questions of how to include both the current generation of nuclear power plants and future plants, how to account for the safety at sites with multiple power plants, and how to include other considerations such as power level and the comparative risks of other methods of generating electricity. The safety goals must also be practical enough that they, and their associated risk insights, can be incorporated into the NRC's regulatory processes. This has presented a challenge when considering new nuclear power plant construction in the U.S. The NRC regulations require that new nuclear power plants licensed under 10 CFR 52 be reviewed against the safety goals as part of the design certification and combined operating license process. Currently the NRC and reactor designers are using the QHOs, the subsidiary goals, and PRAs for the plant as a way of demonstrating that new reactors meet or exceed the level of safety provided by the current generation of reactors. However, because the safety goal policy and its implementation do not distinguish between current and new plants, these reviews are not directly addressing the Commission policy statement on advanced reactors [5] indicating the NRC expectation that new reactors will be substantially safer that the current fleet.

In addition to the potential modification to the QHOs and subsidiary goals that addresses the issues created by reactors having different power levels and sites which have differing numbers and types of reactors on a site is needed, a new QHO which addresses the third qualitative safety goal is needed. With the current emphasis in public policy associated with alternative energy sources and the comparative risks and benefits of each, a QHO that provides a way to measure nuclear power's risk compared to the risks of generating electricity by viable competing alternative technologies is needed. As discussed earlier a method of including the public's relative risk aversion to certain methods of generating electricity should also be considered. An expanded safety goal strategy might be better able to support the construction, operation and regulation of the next generation of nuclear power plants in the United States. Such a strategy must accommodate both the current generation of nuclear power plants and future power plants, account for the safety of sites with multiple power plants, and consider other factors such as power levels and the comparative risk of other methods of generating electricity.

The use of Large Release Frequency (LRF) instead of LERF is viewed by many as a more appropriate safety measure because it provides a better measure of total risk to the public and because it is viewed as more technology neutral that LERF. LERF is based upon the conditional probability of containment failure early in a severe accident with the potential to result in offsite early fatalities. LERF arose from light water reactor technology and includes accident scenarios that would likely result if containment failure or containment bypass occurs in the early part of an accident. Accident scenarios that could lead to late containment failure are not included in LERF calculations.

Some advanced light water reactor and non-light water reactor designs such as high temperature gas reactors have effectively reduced or eliminated the likelihood of early containment failure, by use of passive heat removal systems. Because of the improvements in light water reactor technology in the past forty years, the number and likelihood of the scenarios that would lead to early containment failure, such as the interfacing system loss of coolant accidents (LOCAs), have been significantly reduced. This has led not only to lower LERF numbers, but also to a higher percentage of scenarios that lead to late containment failure. For graphite-moderated, gas-cooled reactor designs, the thermal hydraulics of the reactors tend to ensure a much longer accident progression time, thereby also reducing the number of scenarios that lead to large early releases. One proposal that has been advanced is the use of LRF as an alternative to LERF [18]. LRF would be independent of the time at which the release occurs. Using LRF could put all technologies on a more equal footing with respect to the safety goals, but would not have as close a correlation with the early fatality safety goals, as LERF. Additionally, using LRF will prevent the time of the release from adding uncertainty into the safety metric. Although this is not a major concern for light water reactors, it could have an impact for other reactors that have not been studied as completely. Accordingly because of the extensive amount of work that has been done on LERF for light water reactors, it is not considered to be practical to move away from LERF at this time. However, LERF numbers could be revised to capture multi-site and other issues discuss here.

To address these limitations, this dissertation proposes a modification to the QHOs and subsidiary goals that addresses the issues created by reactors having different power levels and sites which have differing numbers and types of reactors on a site. This research will propose a new QHO which compares competing methods of generating electricity.

CHAPTER 3: LITERATURE REVIEW

Research into the development of safety goals has not been extensive, although there has been some work in the area of how to develop safety goals and what are the most effective methods of evaluating conformance to the goals. Many of these efforts have been based on the desire to have some way to assess relative risks or to compare risks with benefits that an activity provides. Little research into general safety goals occurred prior to the 1970s. Much of the subsequent research has focused on the nuclear arena. This review is divided into three general areas. The first area is associated with how to determine what will be measured and how to identify an acceptable level of risk for safety goals. The second area is the evaluation of risks. This research focuses on the measurement of the risk using tools such as probabilistic risk assessment. The third area deals with the development of safety goals in the nuclear industry.

3.1 Development of Safety Goals

Most of this work is based on the development of methods to compare one proposed alternative with another to determine which is preferable, with cost and/or safety (or hazards) being the most common decision criteria. Hazards arise as a consequence of the need to satisfy societal needs and wants. In part hazards can be modified by the choice of a different technology to satisfy the need or by improving the technology to prevent or mitigate the hazard. The NRC does not have the authority to choose between different power generating technologies. However by establishing an appropriate level of safety, the NRC affects the choice. A number of approaches have been used or advocated for determining whether a particular technology is safe enough or is preferable to an alternative technology including professional judgment, cost benefit or cost effectiveness analysis, comparisons with background hazards, and public preferences. Professional judgment is basically how most regulatory agencies and many industries determine if a technology is safe enough. This judgment is usually based on some form of hazards analysis, estimation of failure rates, and expected costs (dollars and other costs) of failures. However this kind of analysis frequently is incomplete, difficult to aggregate, and lends itself to intended and/or unintended bias in the presentation and interpretation of results [21].

Cost benefit or cost effectiveness analysis are the most common methods for comparing hazards/risk of competing technologies. This type of assessment should be based on the entire life cycle of technology. Very few analyses to date have done so. This issue is discussed further in section 3.3. The risk limit can be established on the basis of cost-effectiveness [22], i.e. the limit is achieved when the expense of safety equipment gets too great. Black et al [23] points out, however, that as the total risk reduction approaches the risk involved in production of the safety equipment (a lifecycle cost), the incremental cost per unit of risk reduction will become infinite. Therefore, if a full understanding of lifecycle risk is not included the use of cost effectiveness criteria is not appropriate. One example of the research on setting social and economic criteria for determining acceptable risk [24] based on cost benefit analysis is the work by Niels Lind.

Lind proposed a “time principle,” coupled with a cost utility analysis for setting safety goals. This time principle simply states that for an alternative action to be acceptable the return to the community in terms of added years of life expectance should be more than the cost in total time to accomplish the activity. In this way an efficiency for the activity and the hours of increased life expectancy per hour spent are determined and used to rank activities. A simple cost-utility analysis is used to determine the value of an average persons work year. This utility is then used as a metric to assess the value of life extension. This provides an effective method for comparing alternative risk reducing (or causing) activities, but does do not provide much information on absolute safety goals. A useful concept that was also highlighted in Lind’s work is the difference between “acceptable” risk and “tolerable” risk, which are frequently used as considerations in establishing safety goals. The UK Health and Safety Executive and others [25] also provide various definitions of these terms. Lind defines “acceptable risk” as risk that most reasonable people would accept in view of the benefits associated with the activity producing the risk and “tolerable risk” as risk that is tolerated regardless of the associated benefits. In this way he has effectively defined the two primary ways of looking at establishing a safety goal. One can establish a safety goal (absolute or relative) that does not depend on the benefit (tolerable risk) or a comparative safety goal that does depend in one way or the other on the benefit to society. Lind’s time principle is similar to most of the other proposed methods in that it provides a means to develop a relationship between costs to reduce risk and safety criteria. The unique aspect of Lind’s time principle is that it calculates the threshold based on the value of a local hour of work. This implicitly assumes that as productivity and cost of living increases it is rolled into the cost utility function, thus reducing the sensitivity to local economic factors.

In the area of public preferences, one area that has been investigated is the concept of revealed preferences. Starr and others [25, 26] have looked at direct and indirect assessments of preferences associated with risk, including direct methods such as opinion polls and psychometric surveys of societal perception of risk and indirect assessments based on how people accept risks in their daily lives and passed regulatory decisions. Although these studies help inform the development of safety goals and the appropriate setting of the acceptance criteria, they suffer from two major assumptions: that what existed in the past was accepted then and is representative of what will be accepted in the future; and that society (people) are well informed concerning the nature of the risks. Fischhoff et al [27], has shown that neither of these assumptions is generally valid. Nevertheless, these studies show that risk perception is real and that the public is generally risk averse. Starr found that technologies that are new, involuntary, uncommon, catastrophic (that is produce a large number of deaths in a single accident), man-made, and dreaded (perceived as lethal and unusually unpleasant) are viewed by the public in a particularly risk averse manner. A safety goal evaluation that does not take this into consideration may not be well received by the public.

For the reasons discussed, the next most common category of safety goals (after cost benefit analysis, and the most common for nuclear power) is based on using comparisons with background hazards. In 1967 Adams and Stone [28] proposed a safety goal for individual risk based on the demographic variation of a person’s risk of death per year. Other methods, including the current safety goals for nuclear power in the U.S. are based on a small fraction of the total risk to an individual in a given year. A number of variations on this basic concept have also been put forth. For example, in 1978 Kinchin [29], proposed a criterion based on the risk of death to an individual member of the public be small [10^{-6} risk of death per year] compared with other involuntary risks that the individual would see in a year.

All of the work in this area has been primarily focused on developing the most appropriate (not necessarily correct or accurate, because this is not strictly a scientific effort as discussed in Chapter 1) method for determining the kind of safety goals and how best to set them. The literature does not point to a single direction but the most widely used safety goal strategies seem to have been based on comparisons with accepted risks and comparisons with other technologies.

3.2 Evaluation of Safety Using Probabilistic Risk Assessment

For any of the above methods to be practical an effective method of measuring the risks to the public in a quantitative manner must be available. In 1957 the first comprehensive examination of the consequences of a large nuclear accident, WASH-740, was published by the Atomic Energy Commission [30]. The purpose of the report was to support Congressional discussion of the Price-Anderson Act[6] on the potential hazards of a nuclear reactor accident. WASH-740 estimated the probability of the occurrence of a severe accident at a nuclear power plant as 10^{-6} per reactor per year. By the late 1960s, the concept of probabilistic risk assessment (PRA) was used both in the academic and the practical analysis of complex systems. The papers by Farmer [31] and Starr [26]

[6] The Price-Anderson Act provides for indemnity against nuclear power plant severe accidents in the form of private and government sponsored insurance.

introduced the concept of risk as a union of consequence and probability and risk perception. At the same time General Electric and Du Pont performed the first simplistic probabilistic analysis of plutonium production reactors [32], while the Boeing Company and NASA began using fault tree analysis to evaluate the likelihood of failures in connection with the Minuteman missile, the Boeing-747, and the Apollo spacecraft. These efforts, along with the growing need to better understand the safety of nuclear power, in 1972 led to the commissioning of the Reactor Safety Study [2] by the Joint Congressional Committee on Atomic Energy. The Reactor Safety Study introduced the concept of event tree analysis while simultaneously attempting to include a realistic analysis of offsite consequences. Before the Reactor Safety Study, it was generally accepted that the consequences of a severe reactor accident would automatically be massive. But, using more realistic calculation of radioactive release from the fuel and the effectiveness of containment systems during the accidents as well as the use of realistic meteorological and demographic data, the Reactor Safety Study showed that most accidents would lead to radiation releases that have only small effects on the public. The Reactor Safety Study examined several thousand core melt sequences and sorted them into 38 general sequences, which were assigned to nine broad release categories for the Surry Unit 1[7] plant (the pressurized water reactor that was used for the study) and seven broad release categories for the Peach Bottom Unit 2 plant (the boiling water reactor used for the study). This method of aggregating core melt and containment failure sequences into a small number of release categories for analysis was developed to reduce the

[7] It is common practice for electric generation sited to have more that one generation facility. These are sometimes all one type of generation unit such as coal or nuclear and sometime a mix. To distinguish one plant from another on a particular site they are usually numbered, for example Surry Unit 1 and Peach Bottom Unit 2.

required computational work given the limited computational resources available at the time. Unfortunately, this method is still commonly used today, even though current computational capabilities make it possible to track each sequence through the entire calculation and to analyze the contributions of each sequence individually.

The most controversial part of the Reactor Safety Study was the comparison it made of the risk from nuclear power plant accidents against risks of more well known events affecting the public, such as automobile accidents, airplane crashes, explosions, dam failures, fires, hurricanes, tornadoes, earthquakes, meteorites and industrial accidents. Although these risks are appropriate for use in providing perspective on the size of nuclear plant risk, because they are well know to the public, they are not appropriate for comparison of risk and benefits without regard to the potential benefits associated with the risks (such as the convenience and time saving aspects of commercial air transport). The critics of the Reactor Safety Study believed that, by displaying the risks of nuclear power in this way; the study "prejudged the acceptable level of risk for nuclear energy, especially when the whole fuel cycle is not taken into account. [33]"

Despite its limitations, the Reactor Safety Study highlighted a number of areas requiring additional regulatory attention when reviewing a nuclear power plant, including small break loss of coolant accidents (LOCAs) and human factors. The follow the publication of the Reactor Safety Study and the significant criticism of it many member of the NRC distanced themselves from it. In June 1976 the House Committee on Insular Affairs held hearing on the report. The hearings found that the report seemed to be misleading with respect to the certainty of its conclusions. The committee along with many of it earlier critics focused on the worst possible accident without taking into account the probabilities associated with them compared to less significant accidents [32]. The Commission requested an external peer review, to be completed by a "Risk Assessment Review Group (also known as the Lewis Committee, after Harold Lewis, the group's chair). The report of this group also provided significant criticisms with some of the details of the methods used in the Reactor Safety Study, but also praised the development probabilistic methods for practical analysis [33]. Following the Lewis Committee Report in 1987, the NRC withdrew its support of the Reactor Safety Study restricted its use in reactor regulation.

Subsequent to the 1979 accident at the Three Mile Island Unit 2 nuclear power plant (TMI-2), is was recognized that many of the insights from the Reactor Safety Study have been demonstrated in the TMI-2 accident. The regulatory perception of the potential value of PRA changed accordingly. Throughout the late 1970s and 1980s, the NRC began to use PRA tools to help resolve reactor safety issues, including support for the Anticipated Transient Without Scram (ATWS)[8] and Station Blackout rules. In parallel to this work a few nuclear power plants developed their own plant specific PRAs. These PRAs generally confirmed the insights provided by the Reactor Safety Study [32].

Subsequent to the TMI-2 accident, the NRC undertook a major effort to improve the understanding of severe accident behavior and the release and transport of radioactive material. In 1986, the NRC initiated effort to undertake a re-evaluation of the Reactor Safety Study risk assessment with improved phenomenological methods. The results of this comprehensive risk assessment were issued as "Severe Accident Risks: An

[8]The ATWS rule, 10 CFR 50.62 "Requirements for reduction of risk from anticipated transients without scram events for light water cooled nuclear power plants," was developed using insights from PRA studies, which provided support for not requiring safety grade systems to meet this rule.

Assessment for Five U.S. Nuclear Power Plants," NUREG-1150 [34]. This study used essentially the same fault tree/event tree approach as used in the Reactor Safety Study but with improved phenomenological modeling and a substantially improved approach to the treatment of uncertainties. This study showed that the risks associated with nuclear power plant operation were comparable to, but lower than was predicted in the Reactor Safety Study.

In November 1988, the NRC issued Generic Letter 88-20, "Individual Plant Examinations for Severe Accident Vulnerabilities [35]." This letter required all nuclear power plants in the U.S. to gain a better quantitative understanding of the overall probability of core damage for their plants. Although the NRC did not require the plants to use PRA, they encouraged the plants to do so and, ultimately, 74 Level 1[9] PRAs of varying degrees of detail (representing all of the nuclear power plants in the U.S.) were completed by 1992 [32]. With the availability of PRAs to support risk analysis, the nuclear plants and the NRC started to use risk assessment in more areas. In 1995 the Commission issued its PRA policy statement [36] directing the NRC staff to use risk in all regulatory matters to the extent supported by the state-of-the-art capabilities and data. From that point the NRC has proactively moved forward with the use of risk analysis and particularly PRA. In 1998 the NRC issued Regulatory Guide 1.174 [12] detailing how PRA results can be used in regulatory decision making. And in 2007 the NRC issued Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities [37], which provided the guidance endorsing the American Nuclear Society's and the American Society of

[9] A Level 1 PRA focuses on the assessment of core damage frequency, rather than examining severe accident progression, containment failure and offsite consequences to the public.

Mechanical Engineers' standards for completing PRAs to provide a standard method for risk analysis.

3.3 Previous Reviews and Developments of Safety Goals in the Nuclear Industry

As discussed earlier, the first comprehensive effort to develop an approach to quantitative safety goals for nuclear power plants was the work done by the ACRS in the late 1970s and early 1980s in response to the reviews of the Three Mile Island accident. This work was published as NUREG-0739 "An Approach to Quantitative Safety Goals for Nuclear Power Plants [10]." This work was the basis of the NRC Safety Goals Policy Statement [1], but all of the recommendations and analysis that were developed and discussed were not implemented in the safety goals. NUREG-0739 was the report to the Commission of the ACRS effort, led by the Subcommittee on Reliability and Probabilistic Assessments' yearlong effort to provide an approach to developing possible quantitative safety goals. The report included a review of possible approaches for quantitative risk criteria, a preliminary proposal for a quantitative safety goal, and an evaluation of the goals for several technologies including nuclear power. The process of developing a proposed quantitative safety goal consists of two tasks, setting the safety criteria and using the estimated the risk to determine if the criteria have been met. This report recommends that criteria include:

- Limits placed on the frequency of occurrence of certain hazardous conditions within the reactor.

- Limits placed on the risk to individuals of early death or delayed death due to cancer arising from an accident.
- Limits placed on the overall societal risk of early or delayed deaths.
- An "as low as reasonably achievable" approach which is applied with a cost-effectiveness criterion that includes both economic cost and a monetary value of preventing premature death.
- An element of risk aversion that is applied to infrequent accidents involving large numbers of early deaths compared to a similar number of deaths caused by many accidents each involving one or two deaths.

For each of these criteria a pair of limits was developed, an upper non-acceptance limit and a lower safety goal limit. Compliance with the upper limit would be required for extended operation of the plant; otherwise the plant would need to be improved within a certain period of time that depended upon the severity of the risk. Any risk value lower that the safety goal level would be considered in compliance for the particular criterion. However, risks would be required to be further reduced below the safety goal levels whenever improvements are possible that met certain cost effectiveness criteria. Between the upper, non-acceptance limit and the lower safety goal level, there would be a discretionary range in which case by case consideration of the uncertainties, regional need for power and alternative risk would be required in the decision as to whether the plant should be allowed to operate for an extended time without modifications.

In this report, the levels for these criteria were established based on reducing the likelihood of a hazard state occurring (prevention) and the ability of containment systems to prevent large scale releases (mitigation). The levels for Criterion 1 (frequency of occurrence of certain hazardous conditions within the reactor) was established at less than 1/100 the probability of a significant core damage event per reactor lifetime ($3x10^{-4}$ per reactor year assuming an average lifetime of a reactor of 33 years) for the safety goal limit. Additionally a safety goal limit of 1/300 probability of large scale fuel melt per reactor lifetime ($1x10^{-4}$ per reactor year) and a safety goal limit of 1/100 probability of a large scale uncontrolled release from containment given a large scale fuel melt were recommended. The levels for Criterion 2 (risk to individuals of early death or delayed death due to cancer arising from an accident) was established at less than the probability of delayed death from cancer to an individual due to all reactors at a site of $5x10^{-6}$ per site year for the safety goal limit, and less than a probability of early death due to a reactor accident of $1x10^{-6}$ per site year. These safety goals were recommended because they were significantly below (in some cases 100 times lower) than the sum of all other risks for any age group. For the societal risk criterion the report recommends an expected value of 2 deaths per 10^{10} kWh (approximately the output of a 1200 MWe plant operating at 95% capacity for one year) from delayed cancer deaths and 0.4 per 10^{10} kWh from early deaths associated with accidents. This report also examined several aspects of risk aversion but focused on the issue of the public aversion to infrequent large accidents with a high number of fatalities in comparison to more frequent smaller accidents leading to the same total number of fatalities in the same time period. The risk aversion model used is provided in equation 3.1 in which α is greater than unity.

$$E_{ev} = \sum_{accidents} (Frequency)(EarlyDeaths)^{\alpha} \qquad \text{Equation 3.1}$$

E_{ev} represents the average number of equivalent early deaths per 10^{10} kWh of electricity generated. If α were equal to one, the equivalent cost would be the same as the expected costs. The value for α is somewhat arbitrary, as it is a measure of relative risk aversion to high consequence events and depends on a number of factors including the education and experience of the population. This report chose a value of 1.2 primarily as an incentive to reduce the catastrophic potential of accidents. This modified method of calculating the risk of early deaths associated with accidents is used for the societal safety goal. For all the criteria discussed above a higher "upper non-acceptance limit" was also recommended.

The final criterion is based on the "as low as reasonably achievable (ALARA)" philosophy of cost-effectiveness, to judge whether additional risk reduction is required beyond the level of safety needed to meet the other safety goals or non-acceptance limits. The cost of an improvement would be balanced against the combined cost in economic losses and the averted cancer deaths. For this analysis a value of $1 million per cancer death averted and $5 million per early equivalent (used equation 3.1) death averted was chosen.

As can be seen from the above discussion this was a very comprehensive analysis (even though it was limited by the available risk assessment methods available at the time). In its review of these recommendations the NRC used a number of the recommendations as starting points for what become the safety goals. However, the NRC rejected the concept of a lower safety goal and an upper non-acceptance limit. Basic safety sufficient to continue to operate a plant was inferred by compliance with the current licensing bases of the plants. The safety goals became truly "goals" and the concept of required safety enhancement beyond the current licensing bases (or below the safety goal levels) when improvements would meet the certain cost effectiveness criteria became the Backfit Rule. The concept of risk aversion did not make it into any of the safety goal policy or other regulatory structures. As part of the tests of these goals this report looked at a nuclear power plant, a coal-fired power plant, and an industrial complex. This provided some context as to the usability of the method and acceptance criteria.

Another approach that was actively developed in the 1970s was a method to compare risk from other technologies as suggested in the qualitative safety goals. Inhaber [38] proposed to include in his review not only the risk of operation and accidents but also the other parts of the "life cycle" of everything that went into providing electricity, including production of the facility, mining and processing the fuel, energy storage for technology that did not produce electricity continuously (solar and wind for example), transportation of fuel and material, and wastes products. He proposed this method of "risk accounting" to try to provide a more complete view of the risk inherent in competing technologies. Although this work was much criticized at the time, the general concept of including all of the potential sources of risk associated with the production of a given benefit is now becoming a widely used method. At the time much less information was available and the quality of the data used for the analysis was not good, so many of the results were questioned. Also methods that Inhaber used for determining the risk of fuel production, component fabrication and plant construction were primarily based on the number of person-hours needed to complete a task, such as mining coal or producing steel. These numbers were then used to generate a metric of risk, which in that study were lost person days per Megawatt year based on known industrial hazard rates. Other parts of the life cycle risk included transportation which was calculated using the amount of material that needed to be transported and known transportation hazard rates. Additionally a rough estimate of the risk of operation and accidents was included. Although not a perfect report, because of many of the limitations in data and methods for calculating health effects it was the first major effort that highlighted the need to look at potential health effects of other parts of the fuel cycle. This concept permits a more appropriate way to assess the different technologies. Because different technologies have their risks concentrated in various parts of the life cycle, this kind of method can provide a more complete view of risks to the public health and safety of a technology.

The NRC has itself investigated the best way to update the safety goals. For example in SECY-05-0130 [18] in 2005, the NRC looked at how best to incorporate the enhanced safety expected of new reactors and the issue of multi-modular reactors. In the attachment to the SECY, the NRC staff examined the possibility of modifying the interpretation of safety goals to establish a minimum level of safety that a new plant must satisfy to achieve enhanced safety, as well as, how to account for multiple reactors on a single site. In this review the NRC staff evaluated in a qualitative manner, the possibility of developing a new QHO for new plants and integrating the risks associated with more than one reactor on a single site. The NRC staff, in this evaluation, did not quantitatively analyze how alternatives to the current safety goals could be carried out, but did examine the potential advantages and disadvantages to modifying the safety goals. As part of this effort, the potential advantages and disadvantages of integrating risk at a site (LERF per site per year) for only new reactors and for all reactors (existing and new) were examined. As discussed in Chapter 2, although the NRC staff recommended that the Commission update the safety goals to account for the integrated risk of new reactors on a site the Commission did not accept this recommendation.

Most recently, a number of authors have studied the risks and benefits of nuclear energy as they compare to other methods for generating electricity. An example of this research is work on defining, in a quantitative way what energy sources are "green." Johnson et al [39] looked a number of criteria associated with impacts to the health and safety of the community and environmental impacts associated with the production of electric power. This research found a wide variation in definitions both in terms of what electric energy sources where considered "green" and what criteria are used to make this determination. As Johnson points out many of the green/not green determinations are not based on formal assessment of the impact on safety or the environment. He suggests that the comparison be based on the potential health and environmental impact of different energy technologies, including all aspects of each technology using a common basis. Johnson and other researchers have defined a common benefit, usually the annual generation of a fix amount of electricity (usually 1000 MW for one year) as a common basis for comparison. Johnson suggested that all aspects of the technology be quantitatively calculated by looking at each element of the fuel cycle, generalized as: extraction; transportation; treatment; storage; conversion to electricity; and waste treatment and disposal. For each technology these general elements will be slightly different but will represent the entirety of the fuel cycle. To develop an accurate view of the total health and environmental impact of technologies, it is necessary to develop data to support the analysis. Johnson provides a brief survey of some of the data that are available in the literature.

Krewitt et al, in the 1990s [40] developed the so called "impact method" for evaluating off site consequences of electric energy production and included detailed consequences of SO_2 and NO_2 as well as nuclear plant accidents. However, Krewitt did not fully account for accidents in non-nuclear systems. The most complete example of this kind of extension to Inhaber's work is the effort led by Stephen Hirshberg at the Paul Scherrer Institut, in Switzerland [41, 42]. Hirshberg's work develops a selection process from among alternative energy sources/technologies based on both internal and external cost assessments. The Hirshberg basic premise is that all energy technologies have both internal (production costs) and external (costs associated with environmental damage and health and safety) costs. Hirshberg has proposed a Multi-Criteria Analysis method to support an analysis of various possible decision-making scenarios. He uses three categories of possible costs; economic, health and environment, and social aspects. The first, economic costs, are the internal costs, the second, health and safety (human health and environmental factors such as loss of crops, greenhouse gasses and wastes), are external costs and the third, social issues, is a mix of social reasons a technology might be favored (employment, risk aversion, waste confinement time, etc.). This research, as with a number of other similar approaches, emphasizes the need to take a holistic approach to the risk and benefits of a given technology. However, these approaches are primarily designed to support determining the most appropriate choice, for a new electric power plant, or more generally for a national strategy for an electric power generation mix.

In the next chapter these concepts will be developed into proposed safety goals that will address the primary purpose of the NRC for establishing the safety goals in the first place, determining in a quantitative sense how safe is safe enough.

CHAPTER 4: DEVELOPMENT OF NEW SAFETY GOALS

As was discussed in Chapters 1 and 2, there are a number of limitations related to the current safety goals. These limitations include the inability of the QHOs and current subsidiary goals (as they are presently stated) to take into account the effects of the different off-site consequences caused by reactors of different sizes (and hence different source terms), the inability of the QHOs and the subsidiary goals to distinguish between sites with one reactor and sites with a number of reactors, and the lack of any difference in goals for current reactors and future reactors. Additionally, there is no QHO supporting the qualitative goal of ensuring that nuclear power plant risks are comparable to, or less than, that risk associated with generating electricity by viable competing technologies.

In evaluating safety goals and developing new safety goals the first task is to determine what kind of safety goal is most appropriate for reactor safety. In his editorial in the September 2005 issue of *Science* [43], Donald Kennedy discussed two types of risk decisions: black-and-white or yes-or-no decisions that are made primarily based on a risk benefit analysis, and societal risk comparison decisions. The Commission, in the Safety Goals Policy Statement [1], established the safety goals as societal risk comparisons, i.e. the Commission chose to set as a goal that the risk of nuclear power be comparable to, or less than, some small fraction of the total societal risk, not that the risk of nuclear power be less than its benefit or that the risk be less than some fixed number. In the Safety Goals Policy Statement, the Commission highlighted the limitations of the available methods for accurately measuring risk as one reason for not going further in working toward the use of the safety goals as prescriptive regulations.

As discussed in Chapter 3, the Reactor Safety Study [2] initiated the use of PRA tools for developing an accurate measure of the likelihood and consequence (risk) of reactor accidents. The use and capability of these tools has continued to improve in the more than twenty years since the first publication of the Safety Goal Policy Statement. The tools are now such that the risk of nuclear power plant operation can be measured and predicted as well as, or more accurately than, most of the other sources of risk to the health and safety of the public in the U.S. What is needed now is to structure the use of safety goals so that they more accurately describe how nuclear power risks might negatively affect the health and safety of the public.

Since the Reactor Safety Study [2] first established risk assessment as a practical quantitative method for determining nuclear reactor safety, research has been conducted to demonstrate that quantitative safety analysis can be used to both set and measure risk. However, as discussed in Chapter 3, although the literature contains a large body of work on general risk acceptance and goal setting associated with societal risk what remains is to establish and demonstrate the effectiveness of improved safety goals. Lind [24], and others [44, 45] have looked at methods for developing social and economic criteria for acceptable safety or risk, but as discussed in Chapter 3, the best form of a safety goal has not been determined. For the past thirty years, the nuclear industry has developed both methods for assessing risk and policies that relate QHOs to quantitative measures of safety or risk [46], but only a modification to the QHOs and the subsidiary goals will address the limitations discussed above.

A new QHO needs to be developed to support the qualitative goal of posing risks comparable to, or having less risk than, that of those posed when generating electricity by viable competing technologies. In response to the claim by the nuclear power community that the use of nuclear power does not produce greenhouse gases, particularly carbon dioxide, environmental groups make the argument that not only does the operation of a nuclear power plant produce some greenhouse gases, but also the construction of the plant, and the mining, processing and enrichment of the fuel produces considerable greenhouse gases. This argument harkens back to an analysis done in the 1970s in Canada by Inhaber [38]. As discussed in Chapter 3, this analysis attempted to develop the life-cycle risk of the construction, operation and decommissioning of an electric generation facility and the risks associated with the mining, transportation and use of fuels. The current term for this is the “environmental (or carbon) footprint” of an activity. To develop a true measure of the relative risks posed by generating electricity by viable completing technologies this type of analysis needs to be preformed. A number or organizations including the U.S. Environmental Protection Agency and the U.S. Department of Energy (through the Energy Information Agency) keep detailed data on greenhouse gas and other pollutant generation [47], associated with the generation of electricity. However this information generally only includes the generation of the electricity and not the other lifecycle hazards or greenhouse gas generation.

Because the current NRC qualitative safety goals are based on high level precepts including the fact that “the societal risks to life and health from nuclear power plant operation should be comparable to or less than the risk of generating electricity by viable competing technologies and should not be a significant addition to other societal risks, and that 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 heath, [1]” this dissertation does not propose to modify these goals. Rather, the modification will be done to the QHOs and the subsidiary objectives. A new QHO will be based on a risk benefit analysis using the generation of electricity as a fixed benefit and the risk of providing that benefit by the completing technologies as the comparison.

A number of simple revisions to the safety goals, the QHOs and the subsidiary objects have been proposed, including scaling them to account for reactor power level, combining plant risks on a site, and other methods to account for the limitations discuss above. In this chapter these and more advance methods will be reviewed and improvements to the safety goals will be proposed.

4.1 Structure of Safety Goals

In evaluating the types of safety goals it is important to establish a structure for the different groupings of risk decision methods and to sort them based on how the comparative acceptance criteria are developed. For this dissertation three types of safety goal categories will be used:

1) Absolute goals: These goals are characterized as fixed numbers independent of other relative considerations. An example of these goals is a state highway organization that sets a goal of some maximum fixed number of deaths on the state's highways over a given time period. These types of goals are common in many safety and regulatory organizations and are usually set at unattainably low values (frequently zero). An absolute nuclear power plant safety goal would need be fixed and independent of the reactor type, site location and characteristics, and would not change over time. Possible examples would include absolute values for the subsidiary goals, CDF, LERF, etc. These goals, which have been used by countries such as Britain and Italy, have been set as goal for CDF (10^{-5} per reactor per year in most cases). Other countries, such as France, have set an absolute goal on the probability of cancers due to reactor operation as 10^{-6} per person per year [46].

2) Comparable risk goals: These are goals that relate a particular risk to a specified individual or society. These goals include comparisons between the risk of harm from nuclear accidents to the risk of cancer from the use of tobacco or pollution or from all other causes. Comparable risk goals provide the advantage of being relatively easy to develop and quantify. However, comparable risk goals generally do not account for relative acceptability of a risk by an individual or by society. For activities that are done actively and voluntarily much higher risks are generally accepted than when the activities are involuntary. In the nuclear arena the current QHOs are comparative risk goals, in that they set an acceptable risk based on a small fraction of the total risk. Additionally comparative risks goals

can effectively change as the comparative risks are reduced. For example, in developing the QHOs the NRC evaluated the cancer fatality rate and the accident fatality rate in the United States ($2x10^{-3}$ and $5x10^{-4}$ respectively). However in the ensuing years a reduction in these rates has led to a reduction in the QHO quantitative criteria (as the rates of cancer fatalities and accidental fatalities fell), even though the 0.1% criteria has not changed.

3) Comparable risk-benefit goals: These are goals that are related to the benefit that the activity is providing. One example of this type of goal would be assessing the value (i.e. the savings in time or the difference in cost) gained as opposed to the risk of traveling using a ground vehicle relative to the risk of travel by air. Another example of a comparable risk-benefit goal is the current NRC qualitative safety goal that looks at the benefits of generating electricity via nuclear power and the related societal risks to life and health from nuclear power plant operation and ensures that they are comparable to or less than the benefits and risks of generating electricity by viable competing technologies. Comparable risk-benefit goals as compared to comparable risk goals tend to be more acceptable when used for risks that have a significant amount of risk aversion among the public, because the risks that are being compared are being generated for the same benefit. In the case of travel, the benefit is getting from one place to the other. In this case all options are voluntary and require personal action. Although the use of a comparative risk-benefit goal provide a method of comparing apples to apples (risk from generating a comparable benefit), other risk aversion concerns may still need to be addressed. Methods such as the ones

discussed in Chapter 3 have been developed to address some of these risk aversion concerns. However, a comparable risk-benefit goal explicitly acknowledges that the activity is designed to provide a measurable benefit and the degree to which the benefit is achieved can be factored into the safety goals and their related acceptance criteria. For example, the current NRC qualitative safety goals provide a context for the risk of nuclear power operation, in that the risk should be similar to other technologies for producing electricity. However the QHOs do not consider the amount of electric power a reactor produces. A modification of the QHOs to account for the relative benefit (i.e. the larger amount of electric power produced) would improve the QHOs by making them proportional to the benefit.

Safety goals can also be viewed as being related to individual and societal risks. One point of view is that an individual undertakes an activity after first weighing the risk of such activity against its direct and indirect benefits. This point of view leads to the setting of an acceptable individual risk (safety goal), which is usually defined as the likelihood (or probability) that when an individual is exposed to a given level of hazard or experience a specific consequence results from this exposure. A second point of view attempts to measure the risk associated with a large population (societal risks) for example looking at the risk posed by the increased numbers of cancer deaths due to exposure to a carcinogen in a given population (or the effects of another hazard). Individual risk goals and societal risk goals are defined differently in the NRC Safety Goal Policy Statement. The individual risk goals focus on prompt fatalities (acute effects) from a reactor accident, and are not concerned with long term health effects to the individual or stochastic heath effects (the likelihood of developing cancer later in life). In contrast, societal risk goals focus on cancer fatalities associated with the operation of the nuclear power plant (stochastic heath effects). Although the NRC Safety Goal Policy Statement includes both exposure associated with an accident as well as exposure associated with normal operations, it assumes that NRC's regulations and oversight will insure the risk associated with normal operations and from routine emissions are small compared to the safety goals. Also, although the individual risk is associated with the risk of prompt fatalities and the societal risk is associated with stochastic health effects, both are associated with accident situations. Another difference is that societal risks, in addition to being defining as a small fraction (0.1%) of the overall risks of living in the United States (from accidents or from cancer), is also associated with the risk of the alternative to nuclear power. In addition to not adding much to the overall risk, the societal risk goal requires that nuclear power also be of comparable or less risk than that associated with other viable technologies that can be used to generate electricity, for which there is no current QHO.

4.2 Potential Strategies for Modifying the Quantitative Health Objectives and the Associated Surrogate Risk Measures

The current QHOs are defined as "the risk to an individual in the vicinity of a nuclear power plant, of fatalities resulting from reactor accidents should not exceed one-tenth of one percent of the sum of prompt fatality risk resulting from other accidents to which member of the U.S. population are generally exposed", and "the risk to the population in the area near a nuclear power plant, of cancer fatalities resulting from nuclear power plant operations should not exceed one-tenth of one percent of the sum of cancer fatality risk resulting from all other causes [1].”

Clearly, since risk acceptability is based on other risks that individuals are exposed to, these risk acceptance criteria for the QHOs are relative in nature but do not account for the relative benefit they provide. There were 97,900 accidental deaths in the U.S. in 2003 [46] and there were an estimated 285,000,000 in the U.S. population in 2003, making the individual risk of accidental death 3.4×10^{-4} per person per year. Based on this estimate, the acceptable annual risk level for an individual associated with living in the vicinity of a nuclear power plant would be 0.1% of this or 3.4×10^{-7} per person per year. According to the American Cancer Society in 2003, there were 556,500 cancer-deaths in the U.S [46]. For a population of 285,000,000 in 2003, the risk of cancer-deaths was 1.95×10^{-3} per person per year. Based on this estimate the acceptable annual risk level for the population near a nuclear power plant would be 0.1% of this or 1.95×10^{-6} per person per year. (Note, as discussed above the original QHOs were 5×10^{-7} and 2×10^{-6} respectively.)

As discussed earlier the subsidiary objectives are a CDF of 10^{-4} per reactor per year and a LERF of 10^{-5} per reactor per year. As part of the original work of developing the safety goals, a number of studies were completed and were validated by numerous PRAs since then, that demonstrate that, for light water reactors source terms[10] for up to

[10] A source term is defined as amount, type and timing of radionuclide release as part of a reactor accident. It is generally reported as release fractions by isotope group or class. The source term is dependent on a number of parameters including the reactor power level, operational history and the release paths.

1000 MWe, meeting the subsidiary objectives for CDF and LERF will result in meeting the QHOs with substantial margin [32].

A number of revisions to the safety goals, the QHOs and the subsidiary objects have been proposed, including scaling them to account for reactor power level, combining plant risks on a site, and other methods to account for the limitations discuss above. These and other possible alternative goals could include:

- Modifying the current QHOs and subsidiary objects to scale them based on electric power level of the reactor (based on benefit they provide)
- Modifying the current QHOs and subsidiary objects to scale them based on thermal power level of the reactor (based on hazard)
- Maintaining the current QHOs for operational reactors, but using a site wide safety goal for new plants, both on new sites and on sites with operational plants
- Use a site wide QHO and subsidiary objectives for all the plants on a site
- Use a site wide QHO and subsidiary objectives for all the plants on a site and scale the goals based on reactor power (either based on electric or thermal power)
- Develop a new QHO for evaluating nuclear power plant risk in comparison with the risks of generating electricity by viable competing alternative technologies.

In the statement of considerations for the Safety Goals policy statement [1], the Commission outlined its reasons and rationale for establishing the safety goals. In this discussion, the Commission states that the risk of release of radioactive material from the reactor to the environment as a result of nuclear power plant operation is its concern. As discussed earlier, the current QHOs and subsidiary objectives do not accurately meet this intent because the risk of release is dependent on the source terms for the reactors and the number of reactors that could potentially affect a given population. The determination to not include a factor to account for the source term or for how many reactors that might be on a particularly site was made primarily for the sake of simplicity at a time when measuring accident risk of a single plant was a significant challenge. However the tradeoff between simplicity and more accuracy needs to be considered going forward. Today, although simplicity in the use and communication of a safety goal should continue to be a consideration, the tools for conducting evaluations of the QHOs and subsidiary goals have improved considerably.

Since the subsidiary objectives map with some margin to the quantitative goals (at least for light water reactors) the simplest way to modify the subsidiary objectives would be to weight them based on power level. Thus, if we use a reference power level of 1000 MWe, calculations for any power level above or below that would then require only that a simple scaling factor be applied. (Since the QHOs make no assumption on the power level of the plant, the current QHO will remain as a safety goal acceptance criteria of the probability of accidental death of 3.4×10^{-7} per person per year, a risk of cancer-deaths of 1.95×10^{-6} per person per year[11], the subsidiary goals of CDF of 10^{-4}, and a LERF of 10^{-5} will be scaled). So if the power level were 1200 MWe (i.e. an increase of 20% to the 1000 MWe), applying the increased 20% scaling factor would provide a CDF of $10^{-4}/1.2$, a LERF of $10^{-5}/1.2$.

Alternatively the subsidiary objectives could be scaled using thermal power instead of electric power. This is more appropriate since the hazards associated with a

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[11] Based on the 2003 data. These data were used because they are the most up to date available based on the 2000 census and National Institute of Health data. When the 2010 census data become available the QHOs should be updated.

reactor accident are due to the release of radionuclides into the environment and the offsite consequences of this release which are related to the core inventory, which is based on thermal power not electric power. This would scale nearly linearly for light water reactors which all have similar thermal efficiencies, but would tend to yield higher (and easier to meet) safety goal thresholds for reactor types such as high temperature gas reactors that have much higher thermal efficiency and, hence, less radiological inventory for a given electric power level. This is appropriate because, for the same benefit, these higher efficiency reactors have less potential to cause harm.

Another modification would be to combine the total power levels (and as such, the inventories) of all of the plants on the site. (Again this would be for the subsidiary goals only as the QHOs are in terms of total hazard regardless of number of plants per site, but the QHOs would now have to be expressed as site QHOs.) This combination could be achieved in a number of ways. The simplest way would be to simply add the power levels of all the plants on the site and adjust the subsidiary objective based on the total power (electrical or thermal). Since the IEs for each plant are independent, the CDF is additive. If LERF corresponds to a similar likelihood of early fatalities for each plant on the site, the values will also be additive. From an accident progression perspective, this assumption is not strictly accurate, but for the most part the plant IEs will come randomly and not at the same time in one plant as they do in another plant. However, for the individual, risk (prompt fatalities have a threshold effect) that could be significantly increased if radiation from more than one plant was to expose an individual. Again the likelihood of this scenario is low because if more than one plant has a severe accident at the same time due to the same cause, the weather conditions will most likely not expose the same individuals. Additionally it would be likely that if the first plant accident were to occur, the population would have evacuated before they could be exposed to the second and subsequent accidents.

The dependencies between plants will primarily affect the probability that there will be IEs that will lead to joint plant accidents in addition to single plant accident sequences for each plant. These joint plant accidents will need to be accounted for if we are to accurately aggregate the risks of all the plants at a single site. In some cases the plants share some systems (diesel generators, the switchyard, the ultimate heat sink, etc.). In some of the new plant designs (the modular high temperature gas reactor (MHTGR) or the Pebble Bed Modular Reactor (PBMR) for example) that have been proposed, they share a larger number of systems. In all cases they are subject to the same external events (including fires, earthquakes, hurricanes, floods, etc) at the same time.

If we wish to add the power levels of plants on a single site, we would have to account for the frequency that an accident sequence would simultaneously produce an offsite consequence from more than one of the reactors on the site. However as discussed above the frequency of multiple unit accidents is probably much lower than that of individual unit events and any incoherence in the releases between multiple units would have a significant effect in reducing the non-linear aspect, a change in wind direction would expose a different population. The earlier alert would also significantly improve the likelihood that the population would have evacuated in advance of the second unit failure. This concept will be explored further later in this chapter and in Chapter 6.

Other issues that complicate the development of a modified surrogate set of safety goals include the potential differences in the source terms based on the kinds of reactor.

If all of the reactors on a given site were of the same general type (light water reactor for example) this would not be a concern. However, if one or more of the reactors has a significantly different set of source terms, a correction factor would need to be included to correct for the difference in its offsite consequences. Alternatively, using the current concept of LERF or LRF, the definition of large release should have a common basis implying a level of potential for early fatalities. Thus, small reactors would need a very large release fraction to be counted as a LERF or LRF whereas for a large reactor the release could be less. In some countries this has been done by redefining LERF to only release that which would expose a person at the site boundary to a particular dose. Additionally, the offsite population density and appropriate emergency planning must be factored in. This is not currently considered in the subsidiary objectives, and to treat different plants at different sites equally in terms of total societal risk, there should be a correction factor based on off-site population density and emergency planning.

Finally the issue of the non-linearity of early fatalities needs to be considered. For sites with more than one reactor, accidents that affect multiple reactors could produce significantly higher offsite doses that could significantly affect the offsite consequences of the accident. This is due to the threshold effect of early fatalities.

As discussed in Chapter 2 and above, one of the limitations of the current subsidiary objectives is the use of LERF. LERF is defined as the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of close-in population such that there is a potential for early health effects. However there is no quantitative definition of the magnitude or the timing of release. That is to say, how large is large is not defined nor is how early is early.

Traditionally in the analysis, large and early are defined by how the containment fails, that is to say which accident progression bins a sequence falls in, determines if it is counted in LERF or not.

The kind of release that would be considered a large release should really be defined as a release that could cause prompt fatalities rather than a release of a certain fraction or a kind of containment failure. Although CDF and LERF have been used as surrogates for the QHOs which do not have these concerns, they have also come to be used as a metric for the acceptable level of safety in and of themselves. Although for the reasons discussed above and in Chapter 2, LERF is not an appropriate measure of societal risk as the QHOs we continue to use it and CDF because unlike the QHOs they provide a measure of defense-in-depth balance between prevention and mitigation.

As discussed in Chapter 2 a more appropriate measure of societal risk would be a release of a sufficient source term that could cause prompt fatalities, not restricted to an early time frame. This is one definition that has been proposed for the metric, large release frequency (LRF). The use of LRF along with CDF would preserve the balance between prevention and mitigation while more closely approximating the societal risk of early fatalities associated with the QHO. Additionally LRF would have a broader applicability to designs in which the release is likely to occur over an extended period or which are generally smaller, either because of the power level or because of the failure modes of the fuel, pressure boundary or containment.

The first level of added complexity above the current subsidiary objects is to develop a set of new subsidiary objectives (i.e. the equivalent CDFs and LERF) that comprise the old subsidiary objectives modified by a weighted sum of the total thermal (or electrical) power level of all of reactors on the site. Thus the new (equivalent) CDF for all the plants on a site would be:

$$CDF_{site} = k\sum_{i} CDF_i\left(\frac{Pe(i)}{\eta(i)}\right) \quad \text{Equation 4.1}$$

where CDF_{site} is defined as the equivalent CDF for a site, CDF_i is the Core Damage Frequency for the i^{th} plant on the site, k is a constant that accounts for the assumed power level of a generic 1000 MWe power plant, Pe(i) is the electric power level of the i^{th} plant on the site and η(i) is the thermal efficiency of the i^{th} plant on the site. And

$$LERF_{site} = k\sum_{i} LERF_i\left(\frac{Pe(i)}{\eta(i)}\right) \quad \text{Equation 4.2}$$

where $LERF_{site}$ is defined as the equivalent LERF for a site, $LERF_i$ is the Large Early Release Frequency for the i^{th} plant on the site and k is the same constant. This constant is adjusted to support the new definition of LERF or LRF as discussed above. The goals for CDF_{site} and $LERF_{site}$ will be set at 10^{-4} and 10^{-5} respectively.

Example: two light water reactor plants, one having a CDF of 3.0 x 10^{-5} per reactor per year, an electric power of 750 MWe and a thermal efficiency of 33%, and the second having a CDF of 6.0 x10^{-5} per reactor per year, an electric power of 1100 MWe and a thermal efficiency of 37%, the equivalent CDF for a site would be

$$CDF_{new} = k[3.0 \times 10^{-5} \times (750/.33) + 6.0 \times 10^{-5} \times (1100/.37)] \quad \text{Equation 4.3}$$

If we use k of 3.5 10^{-5} MWt^{-1} (calculated for a single 1000MWe unit with 35% thermal efficiency we get:

$$CDF_{new} = 8.63 \times 10^{-5}$$

where CDF_{new} is in units of per site per year instead of the traditional per reactor per year.

As the total power level of all the plants increases either from power upgrades or from adding new plants to the site, the CDF and the LERF for each reactor on the site would have to be reduced to maintain the same site CDF and LERF. The site CDF and LERF will be fixed at 10^{-4} and 10^{-5}. As discussed earlier the Commission proposed the safety goals to reflect the risk of operation of nuclear power to the population in the area of the plant. Since the safety goals provide criteria for the relative risk of nuclear power as compared to other man-made activities in society in that area, they should be based on the risk posed by the entire site to the population that it will likely affect. Therefore a site risk is the most appropriate way to assess the safety goals, QHOs and modified subsidiary objectives. A more complete modification would be to include additional factors in the equation that would account for the differences in the radiological inventories between plant types, and the effects due to the conditional probability that if one plant is having an accident the others on the site are also having an accident.

4.3 Strategies for Development of Safety Goals Based on Multiple Reactors on a Single Site

The development and application of the Nuclear Regulatory Commission (NRC) safety goal policy statement [1] laid out the Commission's expectations for the safety of a nuclear power plant but did not address the risk associated with current multi-unit sites, potential modular reactor sites, and hybrid sites that could contain current generation reactors, new passive reactors, and/or modular reactors. The NRC safety goals and the quantitative health objectives (QHOs) refer to a "nuclear power plant," but do not discuss whether a “plant” refers to only a single unit, or all of the units on a site. Additionally, the risk of multiple reactor accidents on the same site has been largely ignored in the probabilistic risk assessments (PRAs) done to date, as well as in most risk-informed analyses and discussions.

As discussed in the preceding section there are a number of ways in which the safety goals, QHOs and subsidiary objectives can be modified to represent site risks. The equations discussed above assumes that the accident progressions can be added for all the plants on a site (equations 4.1, 4.2 and 4.4). This method will not bound the safety of the site, in that treating the plants as completely independent will under estimate the risk.

$$R_{actual} \geq R_{independent} \qquad \text{Equation 4.4}$$

This is true because most of the accident sequences that make up the risk will affect only one plant at a time. However some of the sequences will affect both plants. If we treat the accidents at the different units as completely independent we will not capture these dependent failures.

To evaluating the dependencies between plants on a given site there are several approaches that might be followed. Fleming [48] proposed to develop level 3 PRAs for each plant and an additional level 3 PRAs for “multi reactor accidents” which included

IEs that impacted both units (seismic events, loss of offsite power, tornado and wind, external flooding, etc.) and initiating events that could impact all units under certain conditions (loss of service water, switchyard events, turbine missiles, etc.). In this approach all of the individual plant level 3 PRAs would also be modified to account for dependencies and common cause events between two or more units. In this way all of the dependences between the plants would be captured and any initiating events that affect more than one event at the same time would also be accounted for. This model includes risk from all possible combinations of plants experiencing common initiating events at the same time.

$$\begin{aligned} R(site) = {} & R(plant)_1 + R(plant)_2 + R(plant)_3 + \ldots + R(plant)_n \\ & + R(plants)_{1,2} + R(plants)_{1,3} + \ldots + R(plants)_{1,n} + \ldots \\ & + R(plants)_{n-1,n} + \ldots + R(plants)_{n-2,n-1,n} + \ldots \\ & + R(plants)_{1,\ldots,n-1,n} \end{aligned} \qquad \text{Equation 4.5}$$

Ideally, joint site wide PRAs that include internal and external events completed to level 3 could be performed, that would include sequences such as joint initiating events, multi-plant dependences, and combinations of source terms. However, few plants have completed level 3 PRAs for even a single unit, and it is unlikely they will develop them, barring a significant economic or regulatory requirement. Hence a simplified method for analyzing these issues is needed. Two possible methods would be to simply use the independence assumption when developing the risk number in the evaluation of the safety goals or use a simplified method to analyze the most significant dependences. Based on this doctoral research, a simplified method is proposed [49] that looks at the most significant initiating events and the most significant dependencies between plants on a site and appropriates the interactions between the plants. For the modification to the safety goal either a full level 3 analysis or the simplified method would be acceptable depending on the likelihood of simultaneous accidents at different units. Although a level 3 PRA would be preferable to the simplified method, based on the work by Fleming and the results of the analysis in Chapter 6, the effect will be minor (on the order of 10% of the overall risk), at least for light water reactors. This is a significant finding of this research. The viability of the proposed modified subsidiary goals is dependent, to a large extent, on the ability to use them in a regulatory environment. If all of the plants need to complete level 3 PRAs to implement the updated safety goals, they will not be practical. This will be discussed further in Chapter 6.

Because all operating nuclear power plants in the U.S. have at least a level 1 PRA with internal and external events included and the new plants being proposed for construction in the U.S. all have at least a level 1 design PRA, these analyses can be used to support the simplified analysis of the offsite consequences of both single and multiple reactor accidents.

The simplified method starts by using the first reactor as a base line. Then, subsequent reactors on the same site are added in an iterative fashion to support the analysis of the risk of operation of the additional reactor and the total site risk. The analysis process consists of the following basic steps.

Step one

For the PRAs for each of the plants that are operational or are proposed to be built, list the dominant sequences (that make up 95% of the CDF and 90% of the LERF). Only the dominant sequences are used, to simplify the analysis. It has been shown [46] that although some information will be lost by using only the dominant sequences, the effect on the final CDF and LERF will be minimal (less than 5%). Since the purpose of this analysis is to determine if there is any effect on the CDF and LERF values, approximation will reduce the effort significantly while not affecting the results of interest. Using this list, as suggested by Fleming, review the internal events and all of the events in the sequences and determine if they will be affected by multi-plant issues (common IEs, shared systems, common cause failures across plants, cascading events, etc.). For plants without level 2 PRAs use the generic containment event trees and plant damage states in NUREG-1150 [34].

Step two

For the base line plant, modify the IE frequencies and basic event failure rates to account for the presence of the next plant being added to the site.

Step three

For the i^{th} plant, modify the IE frequencies and basic event failure rates to account for the presence of the basic plant. This would include the availability of additional resources such as additional sources of back-up power, as well as increased failure rates due to hazards associated with failures at the base plant.

Step four

Develop a multi-unit accident PRA using only the IE and dominant sequences identified in Step one.

Step five

Group the outputs (Level 2 source terms) of the previous steps as follows:

Plant one: Base Plant (unmodified) results

Site: Modified Base Plant results plus New Plant results plus Multi-plant results

For simplicity, the Site results will use the same plant damage states as the basic plant PRA and only the dominant states. The results of the New Plant and Multi plant analyses will be combined in these states.

Step six

Complete a level 3 off-site consequence analysis for the plant damage state of the base plant, and for the combination of the Site damage states.

Step seven

Use the difference between the off-site consequences from step six base plants and the Site off-site consequences to assess the safety of the new plant and the Site consequences for the site wide results.

Step eight

Repeat the process for each additional plant. The basic plant for the new iteration will be the site results from the previous iteration. For modular plants the comparison and evaluation completed in Steps six and seven would be done between the base plant and the completed modular plant, for example, between the base plant and after five iterations for a four unit modular plant.

Using this method the added risk of each new plant to a site can be tracked and justified based on the added site risk. The method proposed here will provide a better estimate of the effects of multi-plant dependences, to support the simple bounding analysis discuss above. This method will be tested as part of the evaluation of the practicality of the modified safety goals in Chapter 6.

For modular plants this is particularly important because the currently proposed plants all include a much larger number of shared components compared to current generation plants. However, for all other units on a site that are not part of a modular set, this level of analysis may not be needed because the number of shared components will be significantly less [19]. Additionally, as will be shown in Chapter 6, the multi-plant accident sequences are much less likely than individual plant accidents. Therefore it is recommended that plant PRAs should be reviewed to assess the potential for interdependencies and these should be minimized to the extent practical in the design. This requirement would be similar to the Individual Plant Assessment evaluation done in the 1990s, where plants were required to assess the possible vulnerabilities to severe accidents and to minimize the potential for them to occur.

Based on the discussion above it is proposed to modify the current QHOs and subsidiary objectives of the safety goals as follows:

- QHO One: The risk to an average individual in the vicinity of a nuclear power plant <u>site</u> of prompt fatality that might result from reactor accidents <u>from any or all plants on the site</u> should not exceed one-tenth of one percent of the sum of prompt fatality risk resulting from other accidents to which members of the U.S. population are generally exposed, and
- QHO Two: The risk to the population in the area near a nuclear power plant <u>site</u> of cancer fatalities that might result from operation of <u>any or all of the</u> nuclear power plants <u>on that site</u> should not exceed one-tenth of one percent of the sum of cancer fatality risk resulting from all other causes.

- Subsidiary Objective One: The total site CDF (as developed with either equation 4.1 using 1000 MWe output equating and the simplified site risk model discussed about or other equivalent method) shall not exceed 10^{-4} per site per year.
- Subsidiary Objective Two: The total site LERF (as develop with either equation 4.2 using 1000 MWe output equating and the simplified site risk model discussed above or other equivalent method) shall not exceed 10^{-5} per site per year.

The 1000 MWe power level was used for the analysis to maintain a tie back to the current safety goals which are not power specific, but are used for plants that range from approximately 600 MWe to 1200 MWe. The 1000 MWe would be an anchor value to relate all of the plants to a common point of reference. Additionally, the plants would have the option of using either a full level 3 site PRA, the simplified site risk model or the independent assumption if they completed the evaluation of the amount of dependences that exist between the plants discussed above.

The two significant assumptions of the above proposed evaluation methods for the new subsidiary goals are the independence assumption associated with the multi-plant issues and the use of thermal power level as a surrogate for source term. The limitations of the multi-plant model are discussed above and in Chapter 6. The limitation of the use of thermal power dependent on the relationship between source term and thermal power level. The primary driver of the health effects associated with a severe accident are the short lived isotopes which are dependent primarily on the core power [50]. Other effects include the fuel burn-up, operational history and release pathways, however these have a relatively minor effect (less than 5%) on offsite dose.

4.4 Development of a Comparative Risk-Benefit Quantitative Health Objective for Electric Generation

As was discussed in Chapters 1 and 2, there is no QHO for the qualitative safety goal associated with the risk of generating electricity by other viable competing technologies. This qualitative safety goal is a comparative risk-benefit goal so it is appropriate that a risk-benefit QHO be developed to support it. In the U.S., other viable competing technologies for providing the benefit of large electric production are coal, natural gas, wind turbines, and solar. To develop a comparative risk-benefit QHO for alternative methods of generation of electricity a number of steps must be completed. First a method for comparison must be established, second the metrics for comparison must be determined and finally an acceptance criterion must be develop based on the alternative viable competing technologies discussed above. First we will examine the possible methods for comparison. There are a number of possible choices, but we will focus this discussion on the two most common as discussed in Chapters 2 and 3, cost-based methods and utility-based methods.

4.4.1 Strategies Using Cost-Based Methods

In the area of using risk information for decision making there are a number of methods for evaluating risks from multiple sources. Some of these techniques are based on economic analysis, in which case alternatives are compared based on the direct economic costs and monetized benefits and monetized risk associated with the activity. These approaches to decision making are based on assigning monetary values to the net benefit as they relate to the amount of risk production or reduction and determining whether such risk gains or mitigation are appropriate, desirable and/or practical. These economic methods include cost-benefit analysis, value of money analysis, cost-effectiveness analysis, and risk-effectiveness analysis. Of these the most common is cost-benefit analysis. Cost-benefit analysis is used in certain NRC regulatory analysis, such as the back fit rule. In the back fit rule the cost is the cost of implementing a safety improvement to a plant and the benefit is an estimated improvement in safety as measured by reduction in the frequency or consequence of an accident with negative health effects converted to a dollar figure.

All economic analysis including cost-benefit analysis used in safety analysis is based on establishing risk acceptance criteria that place an acceptable monetary value on human health and ultimately human life, in addition to the other costs and benefits assessed as part of the analysis. While a difficult and controversial practice, it does provide a rational method for evaluating policy decisions such as the safety goals. This key aspect of placing a value on life has been evaluated by several researchers [51]. Several methods have been used for assessing the value of human life:

1. Loss of a person's gross output of goods and services over remaining life. Sometimes gross productivity is reduced by an amount representing the individual's consumption that is how much he or she will produce less how much they will use. This approach usually gives a relatively small value for life, particularly when reduced by consumption.

2. Value of projected income. The present value of future earnings of an individual is estimated and reduced by an amount equal to discounted consumption to obtain a net value. This method also gives a relatively small value for life.

3. Insurance data. In this approach the value of life is indirectly based on the value an individual puts on his or her own life. General data from insurance companies on these amounts are available, but this approach can be biased by the amount of the premium the insured person is willing to pay.

4. Court awards. This method is based on court awards of compensation to beneficiaries of a deceased person. However, this method is also not without bias. Court awards for valuing human life frequently include a subjective component for "pain and suffering" associated with how the person died. That is to say depending on the cause of death the award will be higher, based on the perceived amount of suffering involved for the dying person and his or her family. Although this might be an appropriate method for legal proceeding, it would significantly bias public policy evaluations.

5. Societal investment. This approach is based on the amount of money a person or society is willing to spend to increase their safety or to reduce a source of mortality. It is sometimes difficult to differentiate between the benefit from increasing the perception or feeling of safety and that of actually reducing the number of deaths. The willingness to pay or willingness to accept methods of valuing human life implicitly includes society's perceptions. These values of life, as well as, injury can be developed based on current regulatory policy.

According to VanDoren [52], because government policies reduce risk of death rather that eliminates specific individual deaths, the correct benefit value is society's willingness to pay to reduce risk. VanDoren argues that, if a regulation would reduce risk by one in one million to everyone in a population of one million, then the risk reduction policy or regulation would save one statistical life. In the average willingness to pay for that risk reduction is \$6 per person, then the value of a statistical life is \$6 million. Using data on wages, economists have estimated people's trade-offs between money and fatality risk, thus established a revealed value of human life. Estimates of average risks of death at work imply that, in 2003 dollars, workers appear to receive premiums in the range of \$600 to face an additional annual work-related fatality risk of 10^{-4} per year [46]. This yields an approximate statistical value of life in the U.S. of \$6 million. The implied value of life revealed by a willingness to pay criterion depends on a number of factors. The acceptable expenditure per life saved for involuntary risk is usually higher than for voluntary risks, as people are generally less willing to accept involuntary the same level of risk that they will accept voluntary. The amount people are prepared to pay to reduce a given risk will also depend on the total level of risk, the amount already spent on safety and the resource of the individual. The amount people are willing to pay may also be determined by asking people how much they would be willing to spend to avoid different risks. However surveys carried out to determine these numbers have been shown to be inconsistent, because individuals have a difficult time in answering questions involving very small changes in their mortality. Graham and Vaupel [53] compared the costs and benefits of risk reducing programs. They showed that the values of a life in the U.S. ranged from \$50,000 to \$8 million (in 1978 dollars).

Some regulatory agencies have explicitly or implicitly set values on human life for the purpose of conduction regulatory analysis of the cost-benefit ratios of potential regulations. The NRC uses a criterion of \$2,000 per person-rem, in its regulatory analysis guidelines [54] (using a simple assumption that 500 rem will result in a fatality this relates to a value of acute loss of life of $1 million). All of the methods discussed above are used in conjunction with other costs and benefits associated with the activities to arrive at a policy decision. In the case of improvements to NRC reactor safety [54] for example, these can include:

- reduction in potential loss of life or injury,
- enhancements to health, safety or the natural environment,
- averted onsite impacts,
- averted offsite property damage,
- savings to licensees,
- savings to the NRC,
- saving to State, local or tribal governments,
- improved plant availability,
- promotion of the efficient functioning of the economy, and
- reduction in safeguards risks.

For other policy considerations, other costs and benefits are often used. However, in most case all of the cost benefit methods use a monetary (usually dollars) metric to evaluate alternatives.

4.4.2 Strategies Using Utility Functions

Because the evaluation of risked-based policy decisions usually involves factors that are frequently intangible or imprecise and it is often desired to avoid monetizing risks, cost and benefits, in these cases non-economic methods are used. Noneconomic analysis methods include those that assign a value to risk and other risk management attributes such as cost and time. These values can be real measurable values, such as the cost of loss of life or property, or values perceived by people to be the consequence or undesirability of the event. These methods include a utility analysis, the exceedance method, the analytic hierarchy process, and a structure value analysis (a particular kind of utility analysis). The most common, non-economic method involves the use of utility functions. Utility function analysis was first develop in the early eighteenth century by Gabriel Cramer as a method to formalize the so-called "St. Petersburg Game" which is a classic problem in game theory that was a turning point in how the value of a gamble was determined (a number of probability and risk analysis methods have their roots in early gambling analysis and game theory). Up to that point the "value" of a gamble was based strictly on the "fair" price calculated by determining the mathematical expected value of the outcome. The St. Petersburg Game has an expected value of infinite, but the subjective fair price for the chance to play the game is not infinite, and turns out to be highly dependent on the subject risk tolerance. Since the expected value of the outcome of the gamble was not the appropriate method to value this game, Cramer develop a utility function that was proportional to the amount of money that could be won up to a certain point and constant thereafter. Later mathematicians generalized the utility function and develop the needed properties. A utility is a numerical expression assigned to every possible outcome a decision maker may be faced with. (Generally in a choice between several alternative prospects, the one with the highest utility is preferred.) Utility functions can be either translated or rescaled without affecting the decision. The utility function u(c) is defined as a modulo linear transformation (a constant factor can be added to the value of u(x) for all x and/or u(x) can be multiplied by a positive constant factor without affecting the decision). Utility function can take any number of forms, the most common of which is an exponential utility of the form:

$$u(c) = -e^{-\alpha c} \qquad \text{Equation 4.6}$$

The utility can be generalized as the measure of worth, satisfaction or preference of an outcome to an individual, group or society. It is a dimensionless number that is sometimes referred to as a "utile." A utility function is a real-valued mathematical function that relates outcomes to a measure of worth or "utile." The amount of risk aversion of a given decision can be measured as the curvature of the utility function u(c). The higher the curvature of u(c) the higher the risk aversion. In utility theory the Arrow-Pratt absolute risk-aversion is defined as

$$r_u(c) = -\frac{u''(c)}{u'(c)} \qquad \text{Equation 4.7}$$

For an exponential utility function $r_u(c)$ is constant, for example for the utility defined in Equation 4.6, $r_u(c) = \alpha$. Other utility functions can be defined as well, for example a logarithmic or power form is also used. Logarithmic utility functions have decreasing absolute risk aversion and power law utility functions have increasing risk aversion. Also of importance is that when defining a utility function it must have an upper bound. Although there are a number of other properties of utility functions these is enough for this discussion. Although we have discussed risk aversion, utility functions can also represent risk seeking behaved (a negative absolute risk aversion).

Structured value analysis is particular type of utility function analysis that is frequently used when dealing with the evaluation of imprecise and intangible values in risk management, to avoid monetizing risk, cost and benefits. Structured value analysis is an approach discussed by Rowe [55] is especially useful when the decision maker must consider multiple diverse risk acceptance criteria. In this approach, the risk management parameters such as the risk, consequences, cost, exposure amount, and the time are each assessed based on a value function (a class of utility functions) and a normalized weight based upon the importance of the parameter. An aggregate of the overall value of all factors is then calculated and is used as an index for decision making. Consider the case in which an evaluation of the capability of some risk reduction or risk control options are based on acceptance criteria characterized by parameters *i* (such as cost or risk). Further, assume that the value functions, associated with each parameter *i*, are u_i. These functions are assigned values by the decision makers depending on some predetermined risk metric and/or their preferences and beliefs, and are frequently weighted to assign an importance W_i. This weight is a subjective value expressed by the decision maker directly or through expert elicitation. Then the linear aggregated weights of all parameters would be:

$$V = \sum_i u_i W_i \qquad \text{Equation 4.8}$$

Rowe discussed other forms of equation 4.8 which depend on the nature of the decision. However, equation 4.8 is adequate for most situations. The value of V for all risk reduction solutions is estimated and the one with the best value (either the highest or the lowest, depending on the situation) is selected. This method also allows for consideration of uncertainties in both u_i and W_i values, by propagating these uncertainties to express the values of V in terms of probability distributions. For the purpose of developing a safety goal the decision will be whether or not the total risk of the production of a given benefit (base load electricity) is acceptable. In order to satisfy the safety goal, the value of the risk function for nuclear power plants needs to be similar or less that the alternative method of production of the same benefit.

4.4.3 Structured Value Function for Safety Goal

The qualitative safety goal is a comparative risk-benefit goal so it is appropriate that a risk-benefit QHO be developed to support it. In the U.S., other viable competing technologies for providing the benefit of large electric production are coal, natural gas, wind turbines, and solar. The QHO should set an acceptance criterion that will limit the "total risk" associated with generating a fixed amount of electricity over a given time period from nuclear power to approximately the same risk associated with the least risky competing technology. The "total risk," would then be a measure of the total life cycle risk including health and safety risks. However, how to aggregate the individual risks into a total risk metric and which risks to include then becomes the question.

Hirschberg has proposed to measure each of a number of negative impacts and use decision theory to select which technology would be appropriate to build. For a QHO however, we need to aggregate the health and safety metrics as well as other potential risks. Although one could potentially include any number of negative health and safety aspects of a technology (and the more broadly effects on the local economy, etc.), the most appropriate risks to include would be the primary areas of regulatory concern. These would include:

- prompt fatalities,
- latent fatalities,

- health and safety effects from life cycle activities,
- economic losses and land contamination from accidents,
- operational land and water usage and
- effects of greenhouse gases (not currently reviewed by the NRC but of significant societal concern).

Other potential factors could be included, such as proliferation risks, but these would not be appropriate for a safety goal. Alternatively one could focus only on prompt and latent fatalities from plant operation and accidents and equivalent risks from life cycle activities. This would be limited to the same kinds of risk that are currently evaluated in the current QHOs.

In this dissertation the parameters in the structured value analysis will be the components of risk associated with the cost to society to generate a given benefit. The benefit will be the production of 1000 MW_e of electric power at a capacity factor of 97% with no restrictions on demand availability (power is available when needed regardless of day or night, weather, etc.). The capacity factors for each of the competing technologies will be used to scale the risks, for example the capacity factor for wind farms varies from 20-40% and of solar from 10-20%. To account for the limitations of a number of competing technologies in the ability to provide electricity on demand, we can adjust their capacity factors, based on the local grid load duration curve. However this adjustment is highly dependent on both the characteristics of the grid where plant is located (including the current distribution of base load to peaking plants, the hourly load demand, and seasonal variations). Thus, the risk profile would be different for different parts of the country. Although this would be appropriate for an evaluation of what plant to build, it is not appropriate for a nuclear plant safety goal for the nation. Therefore, we will use a worst case factor for relating non-base load facilities to base load facilities (40%) and then look at this estimate as part of our sensitivity studies in Chapter 7.

4.4.4 Developing the Appropriate Utility (Structure Value) Functions

As discussed above, we will use structure value functions to develop the threshold for the new QHO. Since we are developing a new quantitative safety goal for comparing the risks of nuclear power to the risk of generating electricity by other viable competing technologies, we must develop utility functions for each of the components we will use in the analysis for each of the comparable methods of generating the same benefit. For each component data needs to be collected and scaled for comparison with a given benefit. For this study, the benefit is the generation of 1000 MWe of electric energy for one year, at an assumed capacity factor of 97% (8.5×10^9 kWh). Risks will be for the entire life cycle of the plant, including raw material and fuel production, component fabrication, plant construction, operation and maintenance of the plant, transportation of the materials, fuel and components, and the decommissioning of the plant. This approach is similar to the approach that Inhaber [38] took in his study of comparable heath risks.

In this dissertation both the risk from normal operations and that from accidents will be studied. The availability of data for the various alternative generating technologies is a challenge for this evaluation, although operational data is somewhat available, accident data are not. Inhaber also faced this issue. Fortunately there is some information on accidents available from the Energy- related Severe Accident Database

(ENSAD): a database of information on severe accidents, with an emphasis on the energy sector, which has been developed by the Paul Scherrer Institute [41, 42]. Because accidents in nuclear power plants that result in significant injuries and/or loss of life are rare, there is very little experiential data on nuclear power plant accident risk. Figure 1 shows the accident data from the ENSAD. In this dissertation this will be the primary source of data used and will be modified with updated information where available. The modifications where primarily to remove German specific data (most of the information was develop for plants in Europe, Germany in particular and update information where more up to date information could be found. One example is updating the accident data to include the August 2009 hydro-electric dam accident at Sayano-Shushenskaya in Russia.

Although more data are becoming available for non-nuclear risks, the values are highly dependent on local conditions (e.g., population density, life expectancy, medical support, etc.). These local variations have been removed where possible to provide a more representative database for OECD[12] countries. Data for this dissertation are taken from health effects studies [42] from Germany and the U.S. [46]. Data for some of the other non-nuclear generating technologies needed to be modified. The data for accidents and life cycle risk in the ENSAD data base for example include data on lignite coal that is seldom used in the U.S. but commonly used in Europe (The U.S. uses anthracite and bituminous coal for electric power production). Additionally natural gas data were

[12] The Organization for Economic Cooperation and Development (OECD) is an international organizations based in Paris, France that provides support for improving all aspects of economic growth and support including the energy sector. Member countries in the OECD sometimes referred to as OECD countries include most developed countries in Europe, North America and Asia, but not developing countries.

modified to remove the effects of liquid natural gas accidents which are more common in the data that in the U.S. because the use of liquid natural gas is more limited in the U.S. Figure 2 provides the data on the loss of life expectancy from the generation of electricity from solar power, wind power, hydropower, nuclear power, natural gas, oil, and coal. The health effects are normalized for a 1000 MWe power plant per year of operation. The effect of the low relative energy density of solar energy can be seen as a relatively high loss of life expectancy when normalized to a 1000 MWe power plant. Figure 3 provides data for the combination of both normal operations and accident. The metric used in these comparisons is months of lost of life expectancy. The metric is a useful way to compare relative heath effects of the first three characteristics used in this dissertation, prompt fatalities and latent fatalities due to operation and accidental causes and life-cycle health effects.

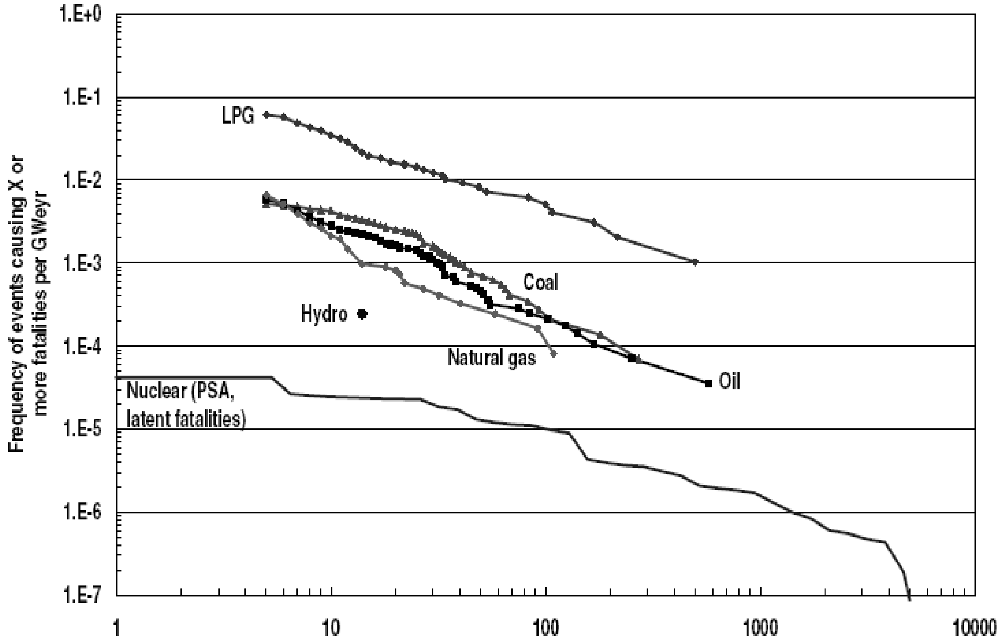

Figure 1: Accident Risk from Alternative Generation Technologies [41]

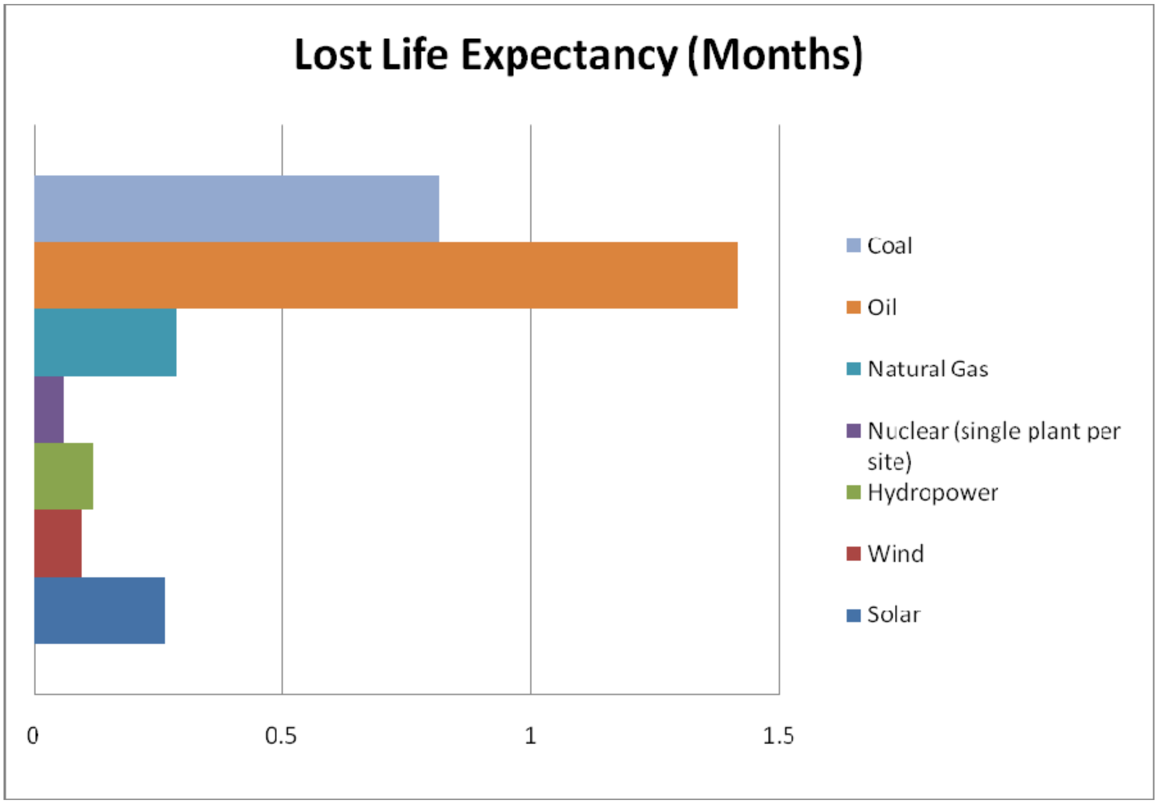

Figure 2: Risk of Normal Operations for Alternative Generating Technologies

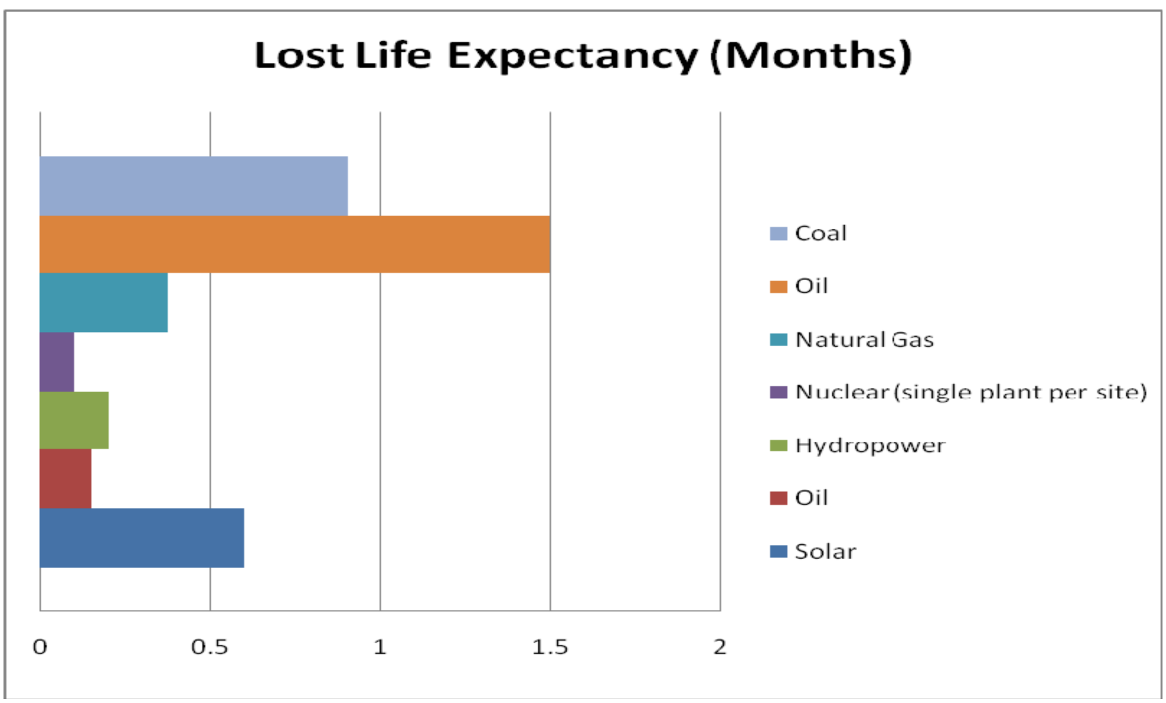

Figure 3: Total Risk from Accidents and Normal Operations for Alternative Generating Technologies

The lost life expectancy is defined as

$$LLE_j = LE_j - LE \qquad \text{Equation 4.9}$$

Where LLE_j is loss of life expectancy due to cause j, and LE_j and LE is the population life expectance with and without cause j. Life expectancy is computed as the weighted sum of the probability of death at a given age as shown in equations, 4.10, 11 and 12

$$LE = \sum_{i-1}^{\infty} i \Pr(i) \qquad \text{Equation 4.10}$$

$$\Pr(n) = h(n)\prod_{i=1}^{n-1}[1 - h(i)] \qquad \text{Equation 4.11}$$

$$h(i) = \frac{\text{number who die in their } i+1^{\text{th}} \text{ year of life}}{\text{number who die in their } i^{\text{th}} \text{ year of life}} \qquad \text{Equation 4.12}$$

By using this metric we can come up with an effective threshold for health and safety for the new QHO either by using loss of life expectance directly or by using it to develop a utility function for these components of health and safety. However this formulization does not account for the issues of risk aversion.

As was discussed in Chapter 2 and 3, the concept of risk aversion is a special case of the more general topic of risk perception. Risk perception refers to people's internal judgment of the frequency of the occurrence and the severity of the consequences of hazards they face. A person's risk perception can be either "risk seeking," viewing the risk of an activity or situation as less risky than an objective analysis would provide, or "risk averse," viewing the risk as more risky than an objective analysis would provide. How the risk is perceived by the people who will be taking the risk or on whom the risk is imposed can be affected by a number of factors. Some of the factors relate to the benefit that the hazard is perceived to provide while others related to the level of dread people feel associated with the hazard. When people expect to benefit from their exposure to a hazard they may be more receptive to expose themselves to the risk. Additionally people are more apt to judge the frequency of a hazard as higher than actual, if they readily remember an occurrence of it or something similar. Also exposures to hazards that lead to severe consequences, even if the frequency of such events is low, attract considerable public and media attention, thus raise the relative dread of the population to that hazard. It is important to recognize that all people, regardless of their role in society, use a speculative framework (based on both their knowledge and emotions) to make sense of the world around them and to form judgments associated with their responses to risk. The more well known and common a hazard is, the more likely people and the overall society will accept a hazard.

Research has shown that risk perception has a real impact on behavior and that the public is generally risk averse (particularly in countries with high standards of living). Several researchers [56] have identified and attempted to quantify the factors that influence people's perception of risk, such as the amount and type of benefit provided by the hazard, control over the exposure, and whether the risk is voluntarily taken or involuntarily imposed. Risks perceived to be voluntary are more readily accepted than risks that are seen to be involuntary or imposed by one group on another. Risks perceived to be under an individual's control are more accepted than risks perceived to be controlled by others. Risks perceived to be familiar are more accepted than risks perceived to be unfamiliar. Table 1 provides some of the most common factors that affect risk perception.

Table 1: Factors Influencing Risk Perception (adapted from reference 44)

Risk Factors	
Delayed	Immediate
Necessary	Luxury
Ordinary	Catastrophic
Uncontrollable	Controllable
Voluntary	Involuntary
Natural	Man-made
Occasional	Continuous
Old	New
Familiar	Unfamiliar

The most fully researched of these factors is catastrophic risk, involving large consequences. These risks are much less acceptable than more common hazards with smaller consequences even if even if the total risk (frequency times the consequence) of the common hazard is much higher. Studies [57] have shown this can be the case even when the reality is that the hazard can only produce moderate consequences but is still perceived to product large consequences. As discussed in Chapter 3 one method for quantifying this risk aversion to catastrophic hazards is to modify the traditional risk equation,

$$R = \sum_{accidents} (Frequency)(EarlyDeaths) \qquad \text{Equation 4.13}$$

to account for an aversion to high numbers of deaths to following equation for equivalent risk [10],

$$R_{ev} = \sum_{accidents} (Frequency)(EarlyDeaths)^{\alpha} \qquad \text{Equation 4.14}$$

In Equation 4.2 α can be adjusted to represent the amount of risk aversion to high consequence hazards. In NUREG-0739, the recommendation was to set α at 1.2; however this was not done based on any particular justification. The effects of the choice of a particular α can be seen in figure 4. In this example the true risk as defined in equation 4.1 is held constant at 0.01 death per year while the consequences are increased from 1 death for the given accident (for illustration proposes this example uses only one accident) to 2000 deaths. In Figure 5 the effects of the choice of α can be seen more clearly. In this example true risk (α = 1.0) is permitted to increase as consequences are increased. As can be seen the choice of α between 1.0 and 1.6 can affect the weighted risk by a factor of more than 95 when the consequences are as high as 2000 deaths. If α were chosen to be 2 or 3, as has been discussed by some researchers, the effective risk at 2000 deaths would 2000 times larger and 4,000,000 times larger which is clearly not a practical model. Although this model of risk aversion to high consequence events is useful for limited situation, it does not provide any information on the other factors that have been shown to affect risk perception. Another method that has been used assigns quantitative levels of added risk aversion to the various risk perception factors. This quantitative evaluation of risk perception is referred to as the risk conversion factor.

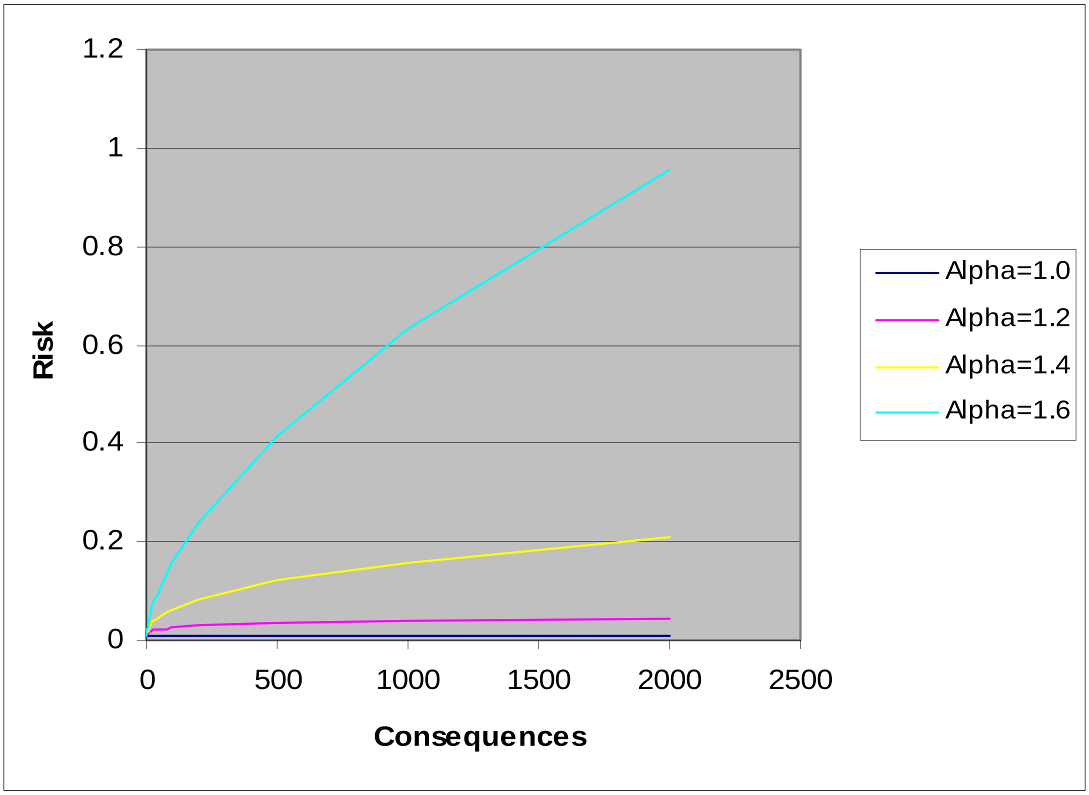

Figure 4: Comparison of the Effects of Risk Aversion Parameter α for Constant Risk

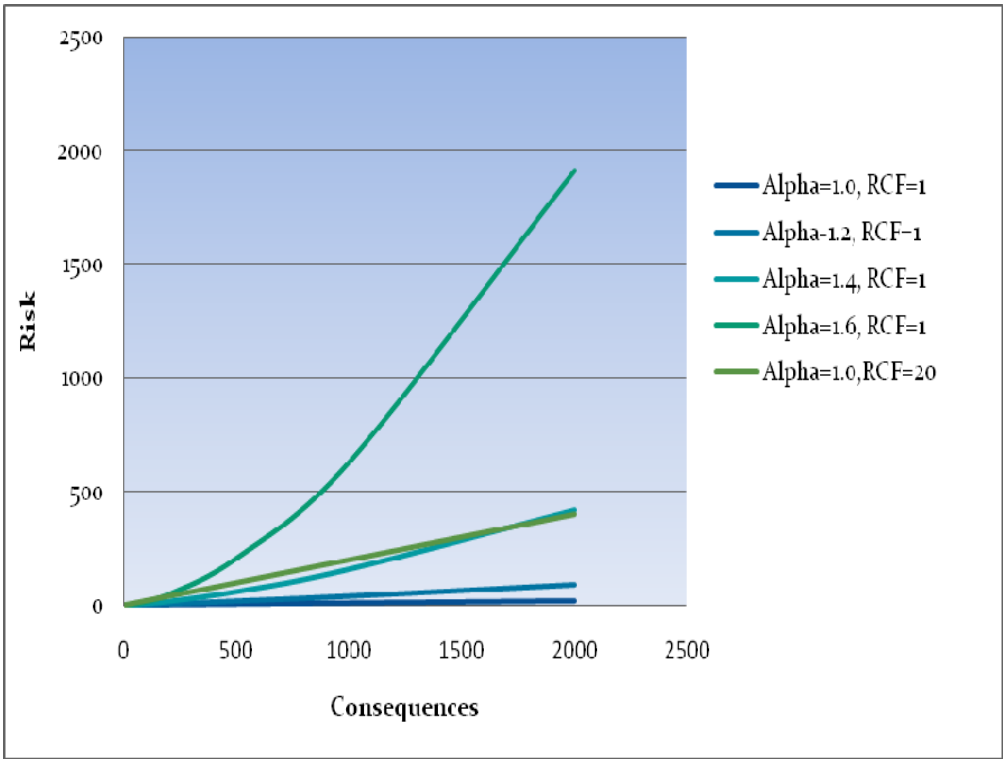

Figure 5: Comparison of the Effects of Risk Aversion Parameter α

Risk conversion factors (RCF) model the relative perception that a hazard will have compared to the alternative. For example an unfamiliar hazard will be perceived as 10 time more risky that a familiar hazard with the same unbiased total risk. There has been some research to support the use of these in a cumulative manner. That is to say that as shown in equations 4.15 and 4.16, the equivalent risk can be determine by product of all appropriate RCFs for a given technology.

$$RCF = \prod_i RCF_i \qquad \text{Equation 4.15}$$

$$R_{ev} = \sum_{accidents} (Frequency)(EarlyDeaths)RCF \qquad \text{Equation 4.16}$$

What is needed is the quantification of the RCF and a determination of which RCF apply to which hazards. Researchers [46] have developed a set of RCFs for some of the most common factors affecting risk perception. Table 2 provides a summary of these RCFs.

Table 2: Risk Conversion Factors (adapted from reference 44)

Lower Perceived Risk	Higher Perceived Risk	RCF
Delayed	Immediate	30
Necessary	Luxury	2-5
Ordinary	Catastrophic	30
Controllable	Uncontrollable	5-10
Voluntary	Involuntary	100
Natural	Man-made	20
Occasional	Continuous	10
Old	New	5
Familiar	Unfamiliar	10

The next step would be to determine which of the factors apply to a particular hazard. One simple way of looking at this phenomenon is to contrast the relative risk of travel by automobile and by airliner. Approximately 42,000 people are killed per year in U.S. in traffic accidents. However, most accidents, even involving fatalities scarcely make the local news and certainly not national news. Accidents associated with commercial air traffic on the other hand are usually in the headlines for weeks.

For this example if we are interested in comparing air transport to driving to a particular location, we would review each of the above RCFs to determine which would affect travel by airlines and travel by car. Although this will be a subjective evaluation, it would seem likely that agreement would be reached fairly quickly that the need in both hazard (crashing) would be immediate, the travel in both cases would be a necessity, both would be voluntary, man-made, continues, old and familiar. However, travel by airliner could be catastrophic and uncontrollable compared to travel by automobile. Therefore the perceived risk of flying compared to driving would be 150 times higher that the statistical risks should show. In fact statistically flying is between 10 and 15 times safer than driving (depending on whether you use miles or hours as a base) [46] while diving is generally perceived to be much safer than flying. The difficulties in using this method include the relative subjectivity as well as, the lack of independence of each of the factors. In the example above the factors of being uncontrollable and catastrophic should not overlap very much. However, for something like nuclear power versus a well known electric power technology such as coal, factors including catastrophic, new and unfamiliar will need to be included in the analysis and these factors tend to be overlapping.

Another limitation of the risk conversion factor model is that the factors are very general and are fixed. That is to say there is no gradation between a technology that is completely new and unknown and one that is only somewhat unknown or fairly new. This is particularly a concern associated with the catastrophic factor. Research [57] has shown that there is a gradation in this factor and it directly affects the risk aversion associated with nuclear power plants. In this dissertation a new form of the risk conversion factor method was developed that uses the aversion power law for catastrophic consequence influence and RCF for the other factors. In this case the effective risk would be

$$R_{ev} = \sum_{accidents} (Frequency)(EarlyDeaths)^{\alpha} RCF^{*} \qquad \text{Equation 4.17}$$

where RCF* would be the RCF calculated in equation 4.15, without the RCF_i for the catastrophic factor. Figure 6 shows some relative risks profiles for equation 4.17. It can be seen that for relatively small values of α, the RCFs will dominate for lower consequences and the power law will dominate as the consequence increase.

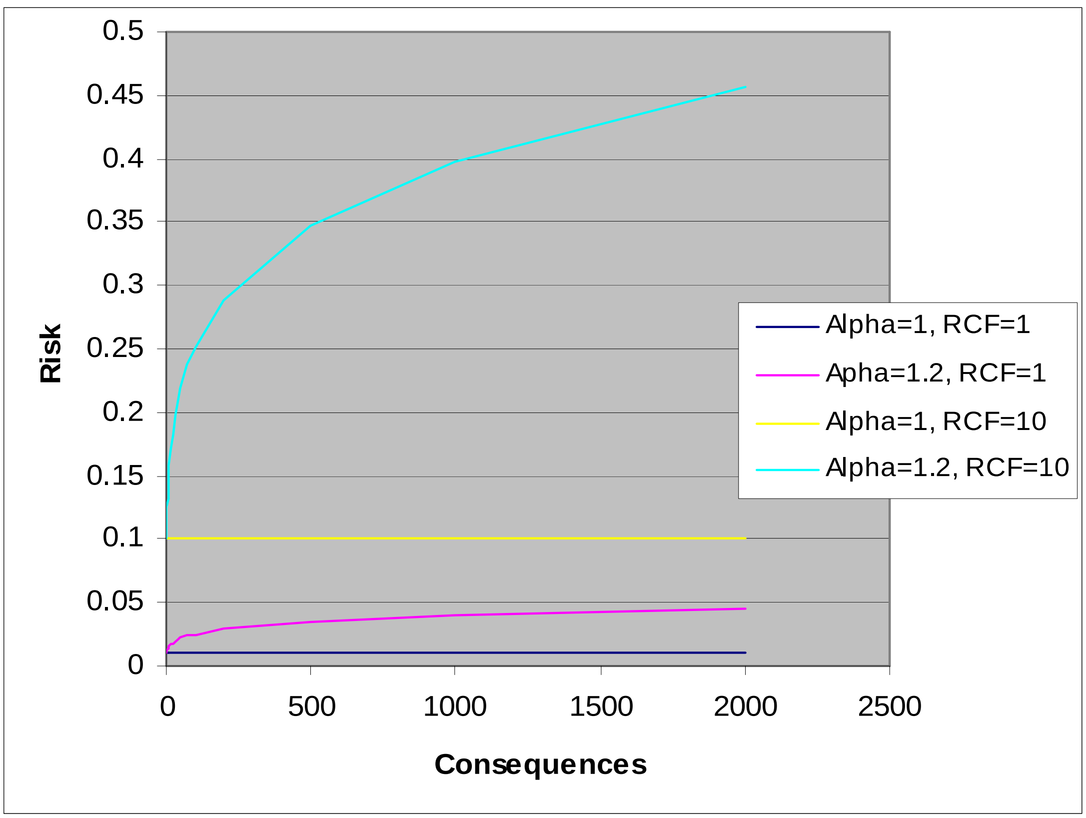

Figure 6: Comparison of the Effects of Risk Aversion Parameter α and RCF for Constant True Risk

Another major concern associated with using any of these models is that there is no standard method to benchmark the results against independent risk perception or public policy measures. One possible method to calibrate one or more of the model parameters (RCF_i and α) would be to use revealed preferences as a benchmark. There has been some research [57] that supports the general uses of the risk conversion factors across technologies; but to assure that the method produces reasonable results (particularly when using RCF that may not be independent) a calibration method is needed.

In the area of risk perception, one area that has been investigated is the concept of revealed preferences. Starr and others [57-58] have looked at direct and indirect assessments of preferences associated with risk, including direct methods such as opinion polls and psychometric surveys of societal perception of risk and indirect assessments based on how people accept risks in their daily lives and past regulatory decisions. The method proposed here will be to look at the societal willingness to pay concept discuss earlier. There have been a number of studies to look at the cost society is willing to pay to improve the safety of certain technologies. This willingness to pay is also found in the NRC cost benefit threshold used in the Backfit rule (currently \$2,000 per person rem of dose averted.). From Tengs' study of life-saving interventions [22] we can see that over the last 40 years a number of regulatory agencies, including the Consumer Product Safety Commission, the Nuclear Regulatory Commission, the Environmental Protection Agency, and other haves used cost-effectiveness and the willingness to pay concept to determine what is and is not an appropriate "cost" for saving a life for different technologies. By using the distribution of willingness to pay for to save a life for the different technologies this dissertation research developed a method to perform a rough calibration of Equation 4.17 for nuclear power and the other technologies associated with the delivery of electric power. To complete the calibration we first calculate the RCF* for each technology (see table 3). Then using the Equation 4.17 and an estimated α of 1.2 develop the estimated equivalent risk for prompt facilities for each technology. Using the information in ENSAD data base, we can determine the number of accidents and the number of fatalities and sum over the accidents to determine the R_{ev} for each technology and compare it to the unmodified risk. This ratio can then be calibrated against the information in Tengs study.

Table 3: Risk Conversion Factors for the Different Technologies

Technology	Delayed /Immediate	Necessary /Luxury	Controllable /Uncontrollable	Voluntary /Involuntary	Natural /Man-made	Occasional /Continuous	Familiar /Unfamiliar	RCF*
Coal	1	1	1	100	20	1	1	2,000
Oil	1	1	1	100	20	1	1	2,000
Natural Gas	1	1	1	100	20	1	1	2,000
Nuclear	1	1	5	100	20	1	1	10,000
Hydro	1	1	1	100	1	1	1	100
Wind	1	1	1	100	1	1	1	100
Solar	1	1	1	100	1	1	1	100

By then using this information we can modify α to get the most appropriate value for the catastrophic factor. In table 4 this analysis is carried out. The final value for α is determine by using an exponential curve fitting.

Table 4: Risk Conversion Factor Calibration

Technology	RCF*	α	$C_{catastrophic}$	R_{ev}/R	Relative cost from willingness to pay	Updated α	Updated RCF*	Updated R_{ev}/R
Coal	2,000	1.2	1.97	3,940	4.2	1.3	2,000	5,400
Oil	2,000	1.2	1.87	3,740	7.3	1.3	2,000	5,120
Natural Gas	2,000	1.2	1.65	3,300	4.1	1.3	2,000	4,236
Nuclear	10,000	1.2	1.89	18,900	34	1.3	5,000	13,100
Hydro	100	1.2	1.82	182	1.7	1.3	100	246
Wind	100	1.2	1	100	1.2	1.3	100	100
Solar	100	1.2	1	100	1	1.3	100	100

Based on this information we can now construct the structured value functions for each of the accident, normal operation and life cycle risk elements of the new QHO. Using Equation 4.8 and each of the first three terms (accident, normal operation and life cycle risk) the total negative effect on society from these factors are determined in term of the unit less “utile.” For this analysis the higher the value the more negative the effect on society from producing the same benefit. Figure 7, provides the updated total structured value function for risk aversion adjusted prompt fatalities, latent fatalities and health and safety effects from life cycle activities.

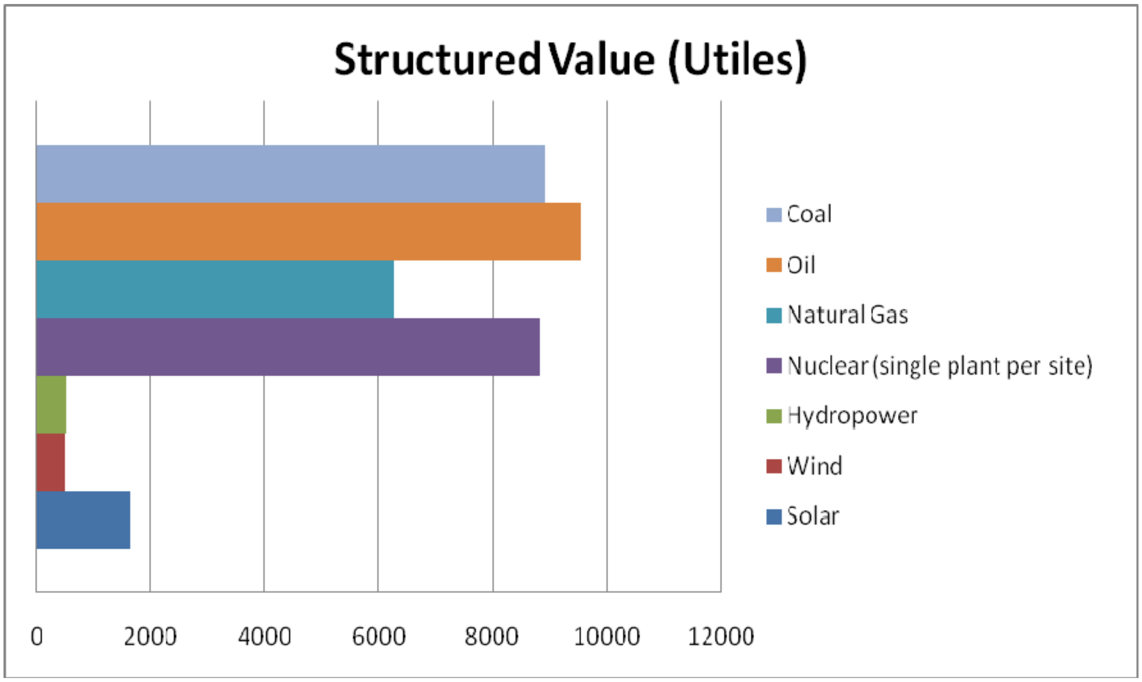

Figure 7: Total Risk from Accidents and Normal Operation for Alternative Generation Technologies (adjusted for risk aversion)

To develop the other parts of the new QHO we need to develop land and water usage, economic losses and land contamination from accidents, and effects of greenhouse gases. The land and water usage figures are available from various data bases, and the land contamination numbers are available from ENSAD. The information on greenhouse gases is somewhat more challenging to obtain as the information needs to include not only the greenhouse gases generated in the operation of the power plants but also in the construction of the facility, the mining of the fuel, transportation of the fuel and construction materials, etc. In this dissertation the effects associated with the generation of CO_2 and other green house gases are included using the total green house emissions. The emission totals in Table 5 are a combination of all green house gases accounting for their equivalent effects on the environment. The structured value for the green house gases was developed using the implied predicted health risk for green house gases used to develop the structured value function. In this case the implied health risk was developed from the current market value of green house gas credits on the European market. Again using Equation 4.8 the rest of the negative societal factor are added in with the first three terms (accident, normal operation and life cycle risk) and then aggregated into the total structured value for all the factors included the risk aversion for accidental deaths. This analysis is presented in Figure 8.

Table 5: Equivalent Green house Gas Emissions of Different Methods of Generating Electrical Power

	Green house gas emissions from power plant (t CO_2 equ./GWh)	Green house gas emissions from rest of chain (t CO_2 equ./GWh)	Green house gas emissions total (t CO_2 equ./GWh)
Coal	679	92	771
Oil	445	104	549
Natural Gas	331	61	392
Hydro	0	4	4
Solar	0	28	28
Wind	0	28	28
Nuclear	0	6	6

As can be seen in Figure 8 when all of the factors are taken into account nuclear has less of an effect on society than coal or natural gas. However, wind and solar are have less impact than nuclear. Although this measure of risk is dependent on a number of assumptions, particularly the amount of risk aversion assumed and the value of reducing green house gas emissions, it provides a measure of the relative risks of providing electricity by the competing technologies. The risk aversion issue is a particularly importation one. If the correction for risk aversion is removed then we would have the structure values as presented in Figure 9.

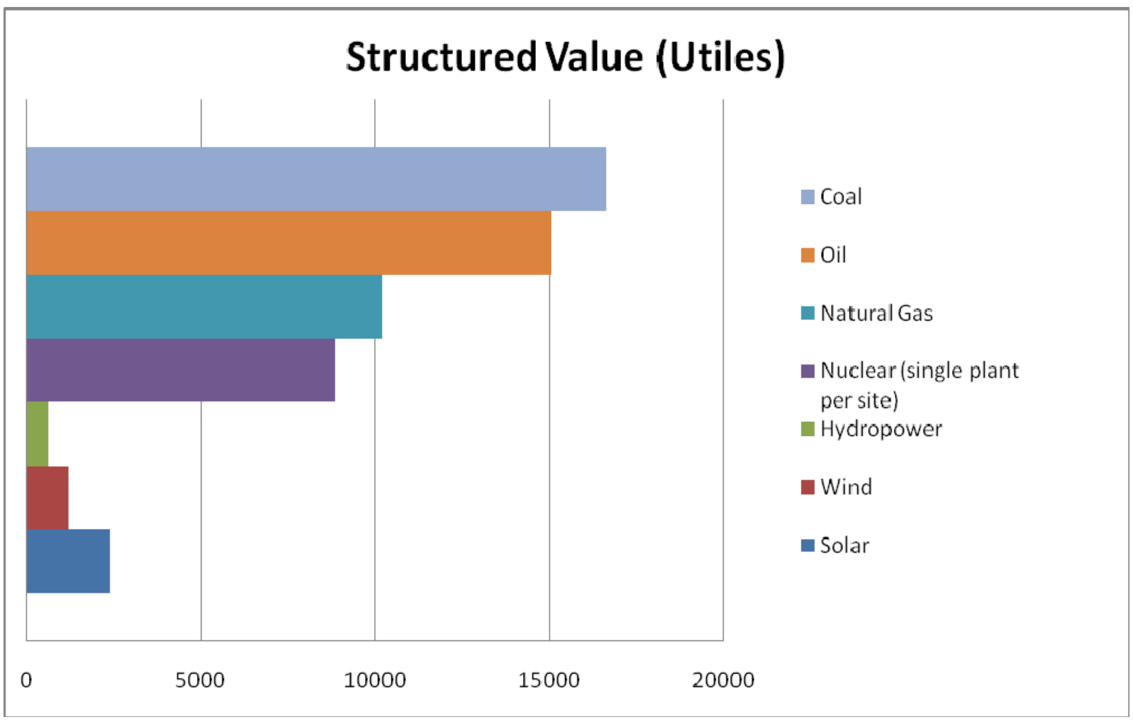

Figure 8: Total Risk from All Factors for Alternative Generation Technologies (adjusted for risk aversion)

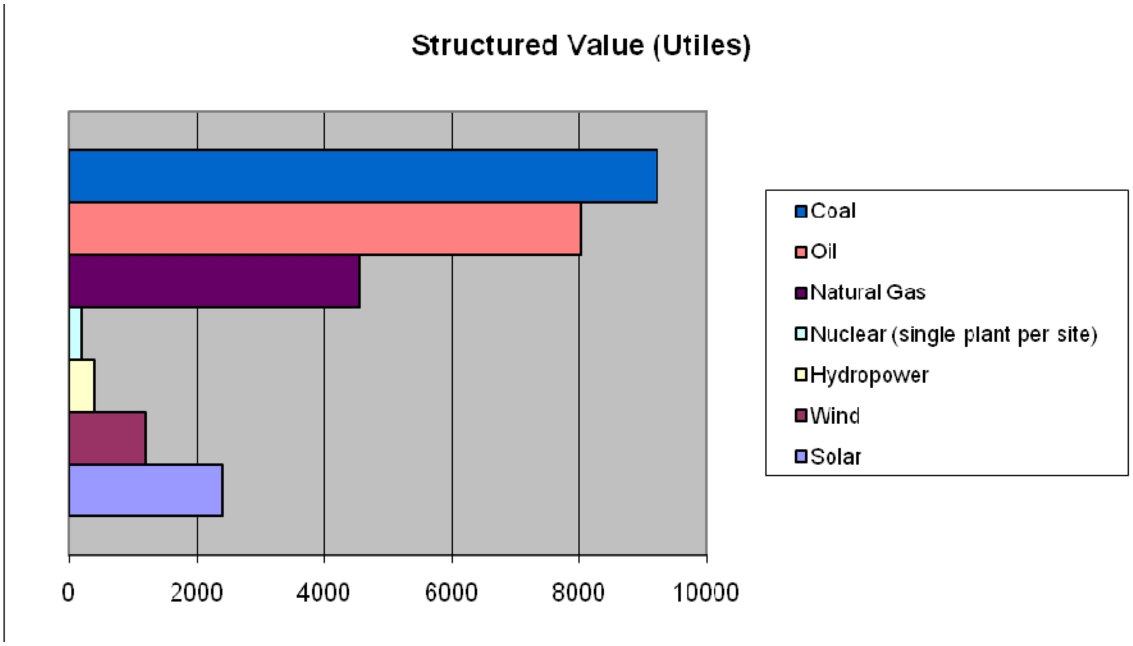

Figure 9: Total Risk from All Factors for Alternative Generation Technologies (without risk aversion)

It can be seen from Figure 9 that nuclear is now clearly the least damaging to society. In addition to the amount of risk aversion to the technologies and the value placed on greenhouse gases there is uncertainty associated with all of the inputs to this analysis. In Chapter 7 these uncertainties will be examined.

4.4.5 New Quantitative Heath Objective

As discuss in Chapter 3, to effectively compare the risk of one technology for generating electric power with any other the all the potential risks associated the generation of the electric power by that technology needs to been evaluated. Some technologies have most of their risks concentrated in the operations (the actual generation of electricity) phase, such as nuclear. Others have a more spread out set of risks, including mining and transportation of fuel, operation and waste disposal, such as oil and coal, while others such as wind and solar have their risk concentrated in the construction phase. Figure 8 provided an aggregated total risk for possible competing technologies. The proposed new QHO is that the societal risk from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing alternative technologies as measured using a total risk model as discussed above. The subsidiary goal for this QHO will be that, on an equivalent base load megawatt hour basis, nuclear should not have a structured value of risk of not more than the competing technologies at the time of licensing. At this point, the most competitive technology would be natural gas generation with a structured value risk of approximately 10,000 utile. If the risk aversion is not used then again the lowest total health and safety structure value for a competitive technology would be natural gas at approximately 4,500 utile, but the nuclear value would be relatively much lower. Additionally without the inclusion of the risk aversion correction nuclear would have a much lower structured value than wind farms, which are not at this time a viable competing technology, but are expected to be so in the near future.

CHAPTER 5: APPLICATION ISSUES

To implement the proposed revised safety goals discussed in Chapter 4 a number of issues need to be evaluated. First, even though the safety goals are not regulation, the Commission has used the safety goals as guidance for setting acceptance levels in a number of regulatory positions and guidance. The safety goals have been used extensively throughout the U.S. nuclear regulatory structure. The areas where the safety goals directly influence the acceptance criteria will need to be reviewed and update, if the proposed revised safety goals are to be implemented. The safety goals are not regulation, but rather are found both implicitly and explicitly as the acceptance criteria; regulatory guidance and review procedures used to determine if the regulations have been met, and include both design and operational evaluations. For example, the regulatory analysis guidelines for backfit analysis, the reactor oversight process and reactor licensing procedures all include acceptance criteria based on the safety goals. On the licensing side, guidance includes procedures for approving reactor license amendments and or licensing new reactors are based, in part, on the ability of the plant to meet the safety goals. Additionally the operational review of reactors including the reactor oversight process and the significance determination process are also based in part on the safety goals.

5.1 Use of the Subsidiary Objectives to Evaluate Proposed Plant Modifications

As discussed above there are a number of areas where the safety goals or the subsidiary objectives are uses to set the acceptance criteria for meting regulatory guidance in the U.S. In August 1995 the NRC adopted the PRA policy statement [36] regarding the expanded use of PRA. The use of PRA was to be "increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the traditional defense-in-depth philosophy." As part of the implementation of this policy the NRC developed the PRA Implementation Plan [59]. The program to development a regulatory method for using PRA findings and risk insights in decisions on proposed changes to a plant's licensing basis[13] resulted in Regulatory Guide 1.174 "An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant-Specific Changes to the Licensing Basis [12]." This Regulatory Guide makes use of the Safety Goals in setting the acceptance criteria for opposed changes. One of the acceptance criteria for proposed changes in this Regulatory Guide is that the "proposed increases in CDF and risks are small and are consistent with the Safety Goal Policy Statement." Regulatory Guide 1.174 uses PRA results in decision making in two ways, by assessing the overall baseline CDF and LERF of the plant and by assessing the changes in CDF and LERF of the plant due to the proposed change. Regions are established in two

[13] As defined in Regulatory Guide 1.174 these are modifications to a plant's design, operation or other activities that require NRC approval. These modifications could include items such as exemption requests under 10 CFR 50.11 and license amendment under 10 CFR 50.90.

dimensions generated by the baseline risk metric (CDF or LERF) along the x-axis, and the change in those metrics (ΔCDF or ΔLERF) along the y-axis and acceptance guidelines are established for each region (see figures 7 and 8). When the calculated increase in CDF is very small, which is taken as being less than 10^{-6} (one percent of the subsidiary objective of 10^{-4}) per reactor year, the change will be considered regardless of whether there is a calculation of the total CDF (Region III). When the calculated increase in CDF is in the range of 10^{-6} per reactor year to 10^{-5} per reactor year, applications will be considered only if it can be reasonably shown that the total CDF is less than 10^{-4} per reactor year (Region II). Applications that result in increases to CDF above 10^{-5} per reactor year (Region I) would not normally be considered. The guidelines are similar for LERF

Therefore plants that meet the current safety goal subsidiary objectives have the freedom to propose greater changes that reduce the safety margin than plants that do not meet the safety goals. In the Regulatory Guide it states that "these guidelines are intended to provide assurance that proposed increases in CDF and LERF are small and are consistent with the intent of the Commission's Safety Goal Policy Statement [1].

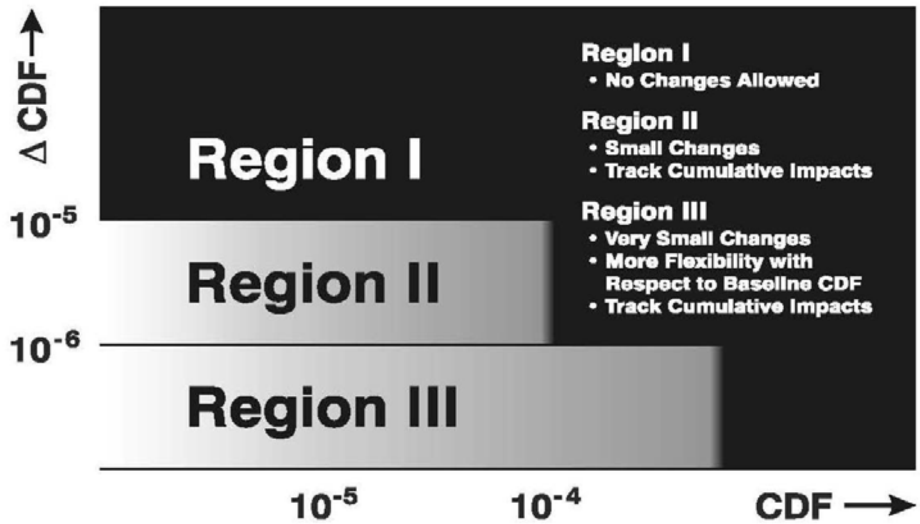

Figure 10: Acceptance Guidelines for Core Damage Frequency (from reference 11)

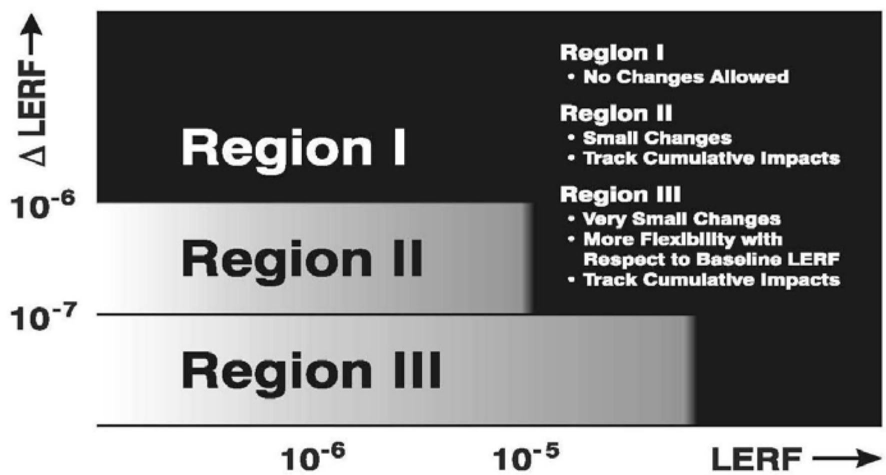

Figure 11: Acceptance Guidelines for Core Damage Frequency (from reference 11)

To effectively implement the modifications to the safety goals and the subsidiary objectives discussed in Chapter 4, this guidance will need to be updated. First the x-axis will need to be updated to account for the new power weighted site objectives for both

CDF and LERF. Next the y-axis would need to be updated to insure that the "proposed increases in CDF and LERF are small." An alternative method would be to establish guidance on a per plant basis within the site criteria for CDF and LERF. Assuming that the total effect of joint failures of multiple reactors on a site is small, then a per plant criterion for CDF and LERF would be simpler to implement. One method for allocating the site CDF and LERF that would maintain the goal of new plants providing significantly lower risk as discussed in the Commission's advance reactor safety policy [5] would be to allocate more of the risk to current plants on a site. A simple method would be to use the follow equations

$$A=\frac{1}{n+10m} \qquad \text{Equation 5.1}$$

$$B=\frac{10}{n+10m} \qquad \text{Equation 5.2}$$

Where n is the number of new reactors on a site, m is the number of current reactors on a site and A is the fraction of the site goal assigned one of the new reactors and B is the fraction of the site goal assigned to the operating reactors. In this way new reactors would be required to maintain their CDF and LERF at one tenth of that of the current reactors on the site. For example if the power weighted site CDF goal is 1.0 x 10^{-4} (per site year) and we have two operative reactors (m) and two new reactors (n) then the faction of the CDF goal for each of the new reactors will be

$$A=\frac{1}{2+10\times 2}=0.045 \qquad \text{Equation 5.3}$$

and the fraction for each current reactor will be

$$B = \frac{10}{2 + 10 \times 2} = 0.454 \qquad \text{Equation 5.4}$$

and the total CDF goals for each reactor would be

New reactor one = 1.0 x 10^{-4} (per site year) x 0.045 = 4.5 x 10^{-6} (per plant year)
New reactor two = 1.0 x 10^{-4} (per site year) x 0.045 = 4.5 x 10^{-6} (per plant year)
Current reactor one = 1.0 x 10^{-4} (per site year) x 0.454 = 4.5 x 10^{-5} (per plant year)
Current reactor two = 1.0 x 10^{-4} (per site year) x 0.454 = 4.5 x 10^{-5} (per plant year).

where the units for the fraction of CDF assigned to each reactor plant would be plant year/site year.

A similar expression could be developed for LERF. However this only deals with the issue of x-axis. For much smaller values of base CDFs (or LERFs) the use of 10^{-5} and 10^{-6} for ΔCDF (or 10^{-6} and 10^{-7} for ΔLERF) no longer makes sense. (As seen in the example above for new plants, the ΔCDF that would be permitted in a modification to the plant would be more than the total plant CDF.) A simple method of dealing with this issue would be to scale the ΔCDF (or ΔLERF) metrics in the same way as the baseline CDF (LERF) was scaled. However this also creates another issue. If the scaling were done for the example new reactor plant above, then the 10^{-5} and 10^{-6} for ΔCDF would become 4.5 x 10^{-7} and 4.5 x 10^{-8}. For most current generation PRAs, the ability to accurately measure changes in CDF is not sufficient to do a reasonable job assessing this level of detail. The uncertainties in the analysis are just too large to make reasonable findings associated with ΔCDF at the 10^{-7} level and below. An alternative would be to use a smaller scaling factor for acceptance criteria that fell below 10^{-5} and 10^{-6}. It is proposed to use the current ΔCDF (and ΔLERF) criteria for all base CDFs greater than

10^{-5} per reactor per year, and to scale by a factor of ten (as opposed to a factor of 100) for all base CDFs smaller that 10^{-5}. In this way a more reasonable ΔCDF (and ΔLERF) can be used without making the measurement accuracy an issue in the regulatory analysis of the proposed change.

It has been argued, particularly by some organizations representing the nuclear industry, that the acceptance criteria (both basic CDF and ΔCDF) should not be changed. That to require newer plants to have and maintain a higher level of safety than existing plants on a site is not appropriate. However, the author would argue that to design and build plants such as the AP1000 with it 2.4×10^{-7} CDF, and then let the plant operators modify, revise and update plant systems, operations and procedures to the point that the plant was operating at a 10^{-4} CDF, would not only violate the intent of the commission's Advanced Reactor Policy Statement, but fail to adequately maintain safety.

5.2 Demonstration of Compliance to the Safety Goals

The NRC staff performs regulatory analysis to support numerous regulatory actions that affect nuclear power plant licensing and operation issues. These include a review of Part 52 applications to determine if they meet the safety goals and regulatory analyses to support additional requirements on licensed facilities under the Backfit Rule [54]. The NRC uses the safety goal policy thresholds for the acceptance criteria for these reviews.

5.2.1 Evaluation of Part 52 License Requests

The Commission originally issued 10 CFR 52 (referred to as Part 52) on April 18, 1989. This rule provided for issuing early site permits (ESPs), standard design certifications (DCs), and combined licenses (COLs) with conditions for nuclear power reactors. In 2007, the NRC published a revision to 10 CFR Part 52, including the requirement for a COL applicant to conduct a plant-specific PRA, and to provide a description of the plant-specific PRA and its results be included within its Final Safety Analysis Report (FSAR). In the review of new plant applications using 10 CFR 52 for both the DC and COL reviews the NRC staff evaluates the design of the proposed nuclear power plant to ensure that the plant will meet the safety goals and that there are not potential accident sequences that might potentially cause an undue percentage of the risk. This is a required part of the review for plants licensed under Part 52. During the review of the plant at the design stage, the NRC staff ensures that the applicant has used the PRA to identify and address potential design features and plant vulnerabilities. By using PRA as a design tool, it is possible to identify and eliminate single or small numbers of component or system failures that could lead to core damage, or large releases. At the COL phase the NRC reviews plant PRAs to ensure that plant specific features of the design have not introduced any new vulnerabilities. Also at both the DC and COL stages of review, the NRC evaluates whether the applicant has demonstrated that the risk conforms to the subsidiary safety goals. With the proposed modifications of the safety goals these comparisons for new facilities would simply require comparison to the new safety goals. None of the proposed new plants in the U.S. individually would have a problem meeting the new safety goals. However, if as is currently proposed for most of the new plants in the U.S., the new plants are to be located on the site of an existing plant, an analysis similar to the one proposed in section 5.1 would be needed. Additionally, since the result of the PRA and how they are used to meet the above criteria must be part of the plant FSAR, the determination of how to apportion the risk among the plants would have to be made at the time the licenses is granted.

5.2.2 Backfit Analysis

The standard against which backfit analysis is done is based in part on a value-impact analysis referred to as a regulatory analysis by the NRC. The regulatory analysis that is performed is designed to meet the intent of Executive Order 12866 and OMB Circular A-4 [60][14]. This guidance provides that regulatory impact analysis shall be performed for all major rules and regulatory actions to ensure that the actions not be taken unless they result in a positive net value to society. As discussed in Chapters 2 and 3, there are a number of possible methods for evaluating what the value to a society will be of any given action. The Executive Order is primarily concerned with imposing new rules and regulations that are costly to industry and the consumers of their products that do not provide significant safety improvements. The NRC Regulatory Analysis Guidelines (NUREG/BR-0058 Revision 4)[54] provides a procedure for conducting this value-impact analysis. The NRC procedure is a multi-step process that includes a screening step and a more detailed value impact analysis. The screening analysis is based on the safety goals, whereas, the value impact analysis includes a number of other criteria

[14] In September 1993, President Clinton issued E.O. 12866, which suppressed E.O. 12291. Sections 1 and 6 of this Order (and the previous Order) direct executive agencies to prepare regulatory impact analysis concerning the need for and consequences of proposed regularly actions. As an independent agency the NRC is not require to comply with these sections of E.O. 12866, however has chosen to do so through the Backfit Rule and the Regulatory Analysis guidelines.

including reduction of public and occupational radiation exposure, enhanced health, safety, reduced impact on the environment, averted onsite impacts, averted offsite property damage, savings to licensees and NRC, improved plant availability, etc.

The screening analysis uses the safety goals to help determine if the substantial additional protection criterion of the Backfit Rule is met. In the screening analysis reduction in CDF for those classes of nuclear power plants for which the proposed new rule or regulation will be applied is calculated. Then, the reduction of the plant's CDF on a per plant year basis is compared to the 10^{-4} criterion. If the reduction in plant risk is greater than the 10^{-4} the screening criterion, a more detailed analysis is performed to determine if the complete "value" to society meets the established threshold. If the reduction in plant risk is less than 10^{-5} per yr, then the screening analysis is failed and unless there are strong engineering or other considerations for continuing further analysis is terminated. For proposed regulatory actions that result in reductions in CDF of between 10^{-5} and 10^{-4} per reactor year, an evaluation of the conditional containment failure probability is looked at. For these situations, if the conditional containment failure probability is less than 0.1 the screening criterion is failed and the analysis is terminated, but if the conditional containment failure probability is greater than 0.1 it is passed and the full value analysis is preformed. The safety goal screening evaluation is used to determine if the proposed backfit may provide substantial additional protection to the public.

As can be seen from the above discussion the subsidiary goal is being used in another way to provide guidance as to how safe is safe enough, in the screening guidance for the backfit rule. If we modify the subsidiary goals as proposed in this dissertation this screening analysis will also need to be modified. One method of modifying the screening analysis criteria would be to scale criteria using the thermal power, number of plants on a site, and whether they are current or existing plants as discussed in Section 5.1. However that section discusses the application of the "how safe is safe enough" criterion for voluntary plant modifications. For the backfit, the regulator is imposing additional requirements that will have already met all appropriate safety requirements and presumably the new safety goals. A backfit is a safety improvement to a plant that is already safe enough, but is warranted because it will provide improved safety at a relatively low cost (the value impact part of the analysis). It is proposed here that the screening criteria be scaled using the thermal power and number of plants on a site, but not on whether the plants are current or existing. This will provide some additional flexibility for plant operators. Because screening analysis calculations are generally done for new generic rules and regulations that will affect a class of plants, this scaling will be somewhat more complicated (the power and site characteristics for each plant in the class of plants that will be affected by the proposed regulation will need to be evaluated) but with currently available generic level one PRAs and modern computing resources, it will not prove difficult. This is a case the independence assumption for site CDF and LERF will be used, because all site to site variations could not be reasonably evaluated. However, the new safety goals could be used effectively and, since changes to "classes" of requirements would most likely affect each type of new plant in generally the same manner, the new plants can be evaluated as a class as easily as any current generation plant.

5.3 Operational Safety Findings

One of the many ways the NRC ensures that nuclear power plants are maintained and operated in a safe manner and in accordance with their license it through a detail assessment and inspection program known as the Reactor Oversight Process (ROP). The ROP is used by the NRC to assess the operational performance of nuclear power plants, by continual evaluation of a number of licensee performance criteria. The regulatory concept of the ROP is to set performance thresholds that reflect the risk and regulatory guidelines that are embodied in existing NRC risk-informed regulatory polices as well as other regulatory requirements. The primary objectives of the ROP are:

(1) That it include multiple levels with clearly measurable thresholds to allow observation and assessment of declining (or improving) performance;

(2) That the thresholds should be risk-informed to the extent practical, and should also include the concepts of defense-in-depth and other existing regulatory requirements;

(3) That risk implications and associated regulatory actions should be consistent with other NRC risk applications, and based on existing criteria where possible (e.g. Regulatory Guide 1.174);

(4) That process should provide for consistency of risk informed indications of performance with performance indications based on existing regulatory requirements and safety analyses to the extent practical;

(5) That the process should be capable of accounting for performance indicated by risk-informed inspection findings;

(6) That the criteria and thresholds should provide sufficient differential to allow meaningful differentiation in performance and limit false positives (which has been generally implemented by allowing an order of magnitude in the risk differential between thresholds);

(7) That sufficient margin should exist between nominal performance levels to allow for licensee initiatives to correct performance problems before reaching escalated regulatory involvement thresholds, and sufficient margin should exist between thresholds that signify initial declining performance and unacceptable performance to allow for both NRC and licensee diagnostic and corrective actions to be effectuated;

(8) That each individual performance indicator (PI) should have its own performance thresholds;

(9) That where appropriate plant-specific design differences should be accommodated; and

(10) That there will be a performance threshold for unacceptable performance sufficiently above the point of unsafe plant operation that the process permits NRC sufficient opportunity to take appropriate action to preclude operation in this condition.

Based on these concepts the ROP was developed. It includes four performance bands, the Licensee Response (or Green) Band, the Increased Regulatory Response (or White) Band, the Required Regulatory Response (or Yellow), and the he Unacceptable Performance (or Red) Band. The general performance characteristics of the bands are:

- The licensee response band is characterized by acceptable performance in which performance attributes and risk indications of individual performance assessment indications (PIs and inspection findings) in the normal range. Performance problems would not be of sufficient significance that escalated NRC engagement would occur. Licensees would have maximum flexibility to "manage" corrective action initiatives.
- The increased regulatory response band is entered when licensee performance is outside the normal performance range, but would still represent an acceptable level of performance. Performance is still considered to be within the objectives of the ROP and the plant technical specifications, but there is indication of declining performance and reduced safety limits. Degradation in performance in this band is typified by changes in risk of up to 10^{-5} CDF or 10^{-6} LERF associated with either PIs or inspection findings. The CDF and LERF threshold characteristics were selected to be consistent with Regulatory Guide 1.174 applications.
- The required regulatory response band involves more significant decline in performance but licensee performance is, in general, still considered acceptable, if marginal. When technical specification limits are exceeded, plants are required to take immediate and effective corrective actions to maintain performance in the band. Degradation in performance in this band is typified by changes in risk of up to 10^{-4} CDF or 10^{-5} LERF associated with either PIs or inspection findings. These threshold characteristics and implied regulatory response are also selected to be

consistent with risk-informed regulatory applications and mandatory actions for regulatory compliance.

- The unacceptable performance band is entered when performance falls below the yellow band threshold. It is typified by changes in performance that are indicative of changes in risk greater than 10^{-4} CDF or 10^{-5} LERF associated with either PIs or inspection findings. Plant performance is considered to be significantly outside the design basis, with unacceptable margin(s) to safety, with an accompanied loss of confidence that public health and safety would be assured with continued operation. Further decline in performance would result in operation in a state inconsistent with the safety goals.

As discussed above, in addition to the setting and monitoring of performance indicators, a process was developed to account for risk-informed inspection findings. The inspections are evaluated using the Significance Determination Process (SDP), to help inspectors determine the safety significance of inspection findings. This process is used to identify those inspection findings that result in a significant increase in risk and thus may have an effect on overall plant risk. The SDP, uses a generic model of plants that includes the number of redundant systems and generic plant vulnerabilities to assess the increase of risk and the amount of time that the plant was exposed to these vulnerabilities to determine the added plant risk. If the level of added plant risk is high a more detailed assessment that may involve NRC analysis may be carried out. The final outcome of the review of the inspection finding is green, white, yellow, or red finding, based on the risk thresholds discussed above for the more general ROP. Each calendar quarter the NRC regional office will review the performance of all nuclear power plants in that region, as measured by the performance indicators and inspection findings. Every six months, this review will be expanded to include planning of inspections for the previous 12-month period. Each year, the final quarterly review will involve a more detailed assessment of plant performance over the previous 12 months and preparation of a performance report, as well as the inspection plan for the following year.

As seen shown above this process is based on Regulatory Guide 1.174 performance thresholds. And as was discussed in section 5.1 the thresholds in Regulatory Guide 1.174 are derived from the subsidiary goals. If we modify the safety goals as is proposed with in Chapter 4, we will need to modify the ROP as well. Since the new safety goals will be plant specific, in that they will be a function of plant power and number of plants on a site, the ROP thresholds will be plant specific as well. This could present some challenges. However, since the SDP and ROP thresholds are also derived based on generic plant characteristics and each plant's specific PIs are tracked separately, his would only become a minor bookkeeping issue.

5.4 Summary

Although these examples of how the new safety goals and subsidiary objects need to be modified are not a complete review of the entire regulatory structure, they give guidance as to how the policy, guidance and acceptance criteria of all of the current NRC regulation and guidance can be modified to account for the proposed modification to the safety goals in Chapter 4. In most cases simple scaling and allocation of risk to particular plants at each site based on the thermal power and assuring the balance between new plant safety and operation plant safety remains appropriate will result in useable site and plant safety goals and can be used within the current regulatory structure.

CHAPTER 6: EVALUATION OF THE PROPOSED NEW SAFETY GOALS

As discussed in Chapter 4 and 5 the proposed new QHOs and subsidiary objectives will provide a more appropriate measure of the health and safety risks to the public and better reflect the qualitative safety goals. In this chapter, the proposed modified and new safety goals will be demonstrated. These tests of the new goals will include sample calculations reflecting the consequences of applying these modifications to several representative reactor site configurations. A number of important assumptions inherent in the current safety goals, QHOs and risk surrogates, including the effect of other risks and potential negative societal effects, such as greenhouse gases (e.g., carbon dioxide), of electric power production, the linear no threshold assumption for health effects of radiation exposure, and the role risk perception could significantly affect this analysis. Uncertainty and sensitivity analyses that will review these potential effects will be discussed in Chapter 7. To test these modifications to the safety goals, several possible reactor designs and configurations including one or more new reactors on a new site, new reactors on an existing site, and small modular reactors, have been evaluated using these new proposed safety goals to determine the goals' usefulness and utility.

6.1 Review of Current and New Plants Against the Safety Goals

As was discussed in Chapters 4 and 5, the current QHOs and subsidiary objectives are used throughout the current regulatory process. Because the QHOs are not used as extensively as the subsidiary objectives, let us first review how these are measured and how the subsidiary objectives are used in their place for most applications. For any given plant the offsite consequences are determined by carrying out a level three PRA or another analysis that approximates it. A level three PRA analysis consists of estimating the frequency of accidents that can cause damage to the reactor core using event tree/fault tree analysis methods to determine the core damage states and the CDF, using the information about the core damage states and the responses of the containment systems to estimate the LERF and the radioactive source terms, and using this information to estimate the consequences to the public and the damages to the environment. The first stage of this analysis is referred to as level one PRA analysis and is carried out using an analysis tool such as SAPHIRE [61] to determine the frequency for each core damage accident sequence. The results of the level one analysis are used as input to level two PRA analysis, which uses similarly tools to analyze the progression of an accident through the containment response to determine the frequency of a release and the amount and type of radioactivity released from the containment. The output of the level two PRA analyses is the LERF and the release information for the release classes. The level three PRA analysis uses the outputs of the level two PRA to estimate the consequences of potential releases of radioactivity based on the characteristics of the release using tools such as the MELCOR Accident Consequence Code System (MACCS2) computer code [50]. The output of the level 3 PRA is the frequency and consequences in terms of health effects (such as prompt fatalities and latent cancers) and land contamination resulting from the release of radioactive material. The analysis for this dissertation uses the SAPHIRE and MACCS2 computer codes to develop the CDF, LERF and health effects results needed to evaluate the proposed new QHO and modified subsidiary objectives. The MACCS2 analysis assumes a representative site. The important parameters/variables required to model the site are the population density/distribution and the site meteorology. The radionuclide inventory, source term (i.e., release fraction, release start time, and release duration), initial plume dimensions (related to the system geometry), and plume heat content. For theses inputs the characteristics of a particular plant were be used. Where available the parameters are taken from public filings with the NRC. Other settings and models necessary for a MACCS2 calculation (food chain model for example) are taken from the NUREG-1150 study MACCS2 input file prepared for the Surry Power Station.

MACCS2 calculates a number of accident consequences, particularly early fatality and cancer fatality, as well as land contamination and other values. The source term, evacuation timing, and sheltering will affect the analysis, so they have been held constant throughout the calculations preformed for this dissertation. The dose conversion factors (DCFs) used in this dissertation are the same as those used in the NUREG-1150.

There are however, two major challenges associated with carrying out these analyses that will need to be dealt with. First the information needed to conduct these analyses for any given operational nuclear power plant or proposed nuclear power plant is considered proprietary and is difficult, if not impossible, to obtain. Some data are available from early reactor risk analyses such as the NUREG-1150 study [34], for operating reactors, and some information is available in the Combined License applications for new reactors. This information is not as up to date as the information available to the utilities and the NRC but should at least provide an effective illustration of the process. The second challenge to carrying out this analysis is the sheer magnitude of the process. A level three PRA is usually a 20-30 man-year process, costing millions of dollars. For these two reasons the analysis in this dissertation will rely on previous level three PRAs to the extent possible.

The plants that are modeled in the dissertation are the Surry operational unit, the AP 1000 advanced light water reactor, and the General Atomic Technologies, Modular High Temperature Gas Reactor (HTGR). Although the HTGR is a much older technology compared to the PBMR or other currently proposed modular reactors, some information needed to carry out this investigation is available for its failure modes, frequencies of failure and release categories. The HTGR, has similar characteristics to other modular reactors, in that it has a much high thermal efficiency, much smaller core inventory and will be built in groups (2, 4 or 8 reactors as a unit) on a site. To develop the off-site consequences for each of the reactor designs to be evaluated the CDF and LERF needed to be developed or obtained. As discussed earlier for the operational plant and the AP1000 this information was developed from previous PRAs. The multiple plant PRA numbers were developed using the new method. For this analysis it was assumed that the multiple plant effects are minimal. For the HTGR there is one available PRA [62] however this PRA was quite crude. To develop the needed CDF and LERF numbers for this PRA this analysis reviewed the IEs and regrouped them fit in the more common categories used today. The analysis used in this PRA assumed minimal confinement integrity and hence the confinement fission product transport models were quite simple and for the most part assumed release of all fission products. Source term estimates were available for all of the operational pant and AP1000 major release categories, but not for the HTGR. These source terms were develop by iteratively using available information on offsite mean doses for thyroid and whole body exposure. For each plant type and each release category MACCS2 calculations were then completed to develop the off-site consequences associated with the particularly sequence.

For the analysis, first we will look at the AP 1000 and the Surry plant with respect to the current and modified safety goals as single plants. Calculating for the Surry plant we find that the CDF is 4.1×10^{-5} per reactor year and LERF is 3.7×10^{-7} per reactor year. For Surry, the individual early fatality risk is 1.7×10^{-8} deaths/yr and the individual latent cancer risk is 2.0×10^{-9} deaths/yr For the AP-1000, the CDF is 2.4×10^{-7} per reactor year and LERF is 1.95×10^{-8} per reactor year. For AP-1000 the individual early fatality risk is 1.37×10^{-10} deaths/yr and the individual latent cancer risk is 3.42×10^{-12} deaths/yr. For the HTGR, the CDF is 6×10^{-7} per reactor year and LERF is 7.1×10^{-5} per reactor year. For HTGR the individual early fatality risk is zero, because the amount of the release is small and the individual latent cancer risk is 2.04×10^{-9} deaths/yr. All of these are well within the current QHOs and subsidiary objectives with the exception of the LERF for the HTGR. As discussed earlier since no postulated release from an HTGR will provide an adequate dose to cause a prompt fatality, none of the releases are really a "large release." However, the LERF number (which is the same as the CDF for this reactor, because the confinement is assumed to fail) is carried in the analysis for completeness. Additional information on the calculations is provided in Tables 6, 7 and 8.

Table 6: Surry Off Site Consequences

Release Categories	CDF	LERF	Individual Prompt Fatality Risk	Individual Cancer Fatality Risk
VB, Alpha, Early CF	1.2 E-6	1.2 E-7	1.71 E-10	8.02 E-11
VB, >200 psi, Early CF	3.4 E-6	3.3 E-8	3.06 E-9	1.6 E-10
VB, <200 psi, Early CF	3.9 E-6	2.8 E-8	0.0	0.0
Late Containment Leak or Basemat Melt-Through	2.9 E-8	6.3 E-8	1.83 E-10	1.56 E-10
Containment Bypass	3.1 E-8	3.6 E-8	1.36 E-08	1.58 E-9
No Containment Failure	4.0 E-5	NA	NA	NA
Total	4.1 E-5	3.7 E-6	1.7 E-8	2.0 E-9

Table 7: AP-1000 Off Site Consequences

Release Categories	CDF	LERF	Individual Prompt Fatality Risk	Individual Cancer Fatality Risk
Intact Containment (IC)	2.2 E-7	NA	0.0	5.11 E-14
Early Containment Failure (ECF)	7.5 E-9	7.41 E-9	7.38 E-12	6.17 E-13
Intermediate Containment Failure (ICF)	1.9 E-10	1.95 E-10	1.74 E-12	2.95 E-14
Late Containment Failure (LCF)	3.5 E-13	1.60 E-10	0.0	6.25 E-17
Containment Isolation Failure (CIF)	1.3 E-9	1.37 E-9	6.23 E-14	2.16 E-17
Containment Bypass (CB)	1.1 E-8	1.12 E-8	1.28 E-10	2.72 E-12
Total	2.4 E-7	1.95 E-8	1.37 E-10	3.42 E-12

Table 8: HTGR Off Site Consequences

Release Categories	CDF	LERF	Individual Prompt Fatality Risk	Individual Cancer Fatality Risk
Large and Moderate primary coolant leaks (Area > 1 in.^{2})	1.0 E-6	1.0 E-6	0.0	8.0 E-14
Large and Moderate primary coolant leaks (.003 in.^{2} <Area < 1 in.^{2})	1.0 E-5	1.0 E-5	0.0	8.0 E-13
Steam Generator Leaks	2.0 E-5	2.0 E-5	0.0	4.0 E-11
Depressurized Conduction Cooldowns	4.0 E-5	4.0 E-5	0.0	2.0 E-10
Total	7.1 E-5	7.1 E-5	0.0	2.408 E-10

To assess these individual plants against the new version of the subsidiary goals their CDF and LERF need to be adjusted for plant power level and efficiency as provided in Equations 4.1 and 4.2. Table 9 provides these numbers. There has been some discussion as to whether or not the proposed CDF and LERF numbers calculated for the AP-1000 plant are realistic. This is because extensive credit has been given in the design PRA for the new passive features used in the AP-1000 design. Although approved by the NRC there is some concern that these features may not be as reliable as they are claimed. Additionally as discussed in Chapter 5, as CDF and LERF numbers get to be as low as the ones calculated for the AP-1000 the uncertainty is called into question. Therefore in this analysis a higher set of values for the AP-1000 reactors is also included. As can be seen in table 9, none of the individual plants will have any difficulties in meeting the power adjusted subsidiary objects.

Table 9: Power Adjusted Individual Plant QHO and Subsidiary Objectives Evaluation

Release Categories	CDF_{new}	$LERF_{new}$	Individual Prompt Fatality Risk	Individual Cancer Fatality Risk
Surry	3.65 E-5	3.30 E-6	1.7 E-8	2.0 E-9
AP-1000	2.86 E-7	2.32 E-8	1.37 E-10	3.4 E-12
AP-1000 x 10	2.86 E-6	2.32 E-7	1.37 E-9	3.4 E-11
HTGR	8.73 E-6	8.73 E-6	0.0	2.408 E-10

6.2 Review of Multi-Plant Sites

As discussed in Chapter 4, sites that have more than one reactor will be subject to some level of interdependency between the reactors. This will be particularly true for modular plants like the HTGR. There have been only a few studies done of this effect [48, 49] and they were briefly discussed in Chapter 4. To provide more context to the assumption made in Chapter 4 that these cross plant effects are small and for this analysis can is assumed to be much smaller than the other multi plant issues, a brief quantitative summary of these studies is provided. The first step in the method proposed by Arndt [49] starts with using the first reactor as a base line, in this case the Surry plant. The, subsequent reactor on the same site for this illustration will be the AP 1000 reactor. As discussed in Chapter 4, for the Surry and AP-1000 plants a list the dominant sequences was developed that included, LOCA, SGTR, LOOP, Transients, ATWS (IS LOCA was not included because of its very small contribution to CDF for both plants). Using this list, a review the internal events and the events in the sequences was completed to determine if they will be affected by multi-plant issues (common IEs, shared systems, common cause failures across plants, cascading events, etc.). In this case the LOOP was the most significant potentially affected.

Then for the base line plant (Surry), the IE frequencies and basic event failure rates were modified to account for the presence of the next plant being added to the site. For the second plant (AP 1000), the IE frequencies and basic event failure rates were modified to account for the presence of the basic plant. This included the availability of additional resources such as additional sources of back-up power, as well as increased failure rates due to hazards associated with failures at the base plant. In this case the diesel generation failure and recovery rates were adjusted to account for the present of additional generators on the site, for example.

Next a simple multi-unit accident PRA was developed using only the IE and dominant sequences identified earlier. The multi results will use the same plant damage states as the AP-1000. The results of the modified plant CDF and LERF and multi plant (both plants having a severe accident from a single IE) CDF and LERF were then calculated. To complete the analysis an off-site consequence analysis was performed for the major contributors similar to the analysis presented in tables 6-8. The results of this analysis show that the changes to the base CDF and LERF for the two plants is very small and the multi unit CDF and LERF were less than 5% of the base plant CDF and LERF (1.9 E-6 and 1.2 E-7). The off-site consequences were similarly small. These results are similar to the results presented in Fleming [48].

Although this analysis shows that the assumptions made regarding the relative independence of reactors on multi-reactor sites, a caution should be included. The analysis did not included reactors with highly integrated operation and systems, such as the proposed modular reactors. This analysis should be reviewed and updated if it is to be applied for these reactors. For the remainder of this dissertation we will assume the independence of the IE and hence the CDF and LERF for reactors on a site.

Using the methods and tools discuss above it is possible to modify QHOs and surrogate risk metric (CDF or LERF) to account for site risk. Based on the discussion above the proposed site QHOs, site CDF and site LERF (based on Equation 4.1 and 4.2) are presented in Table 10 below for a number of possible site configurations.

Table 10: Power Adjusted Site QHO and Subsidiary Goal Evaluation

Site Configuration	Site CDF	Site LERF	Site Risk (Early Fatalities)	Site Risk (Latent Cancers)
Current generation single unit on site	3.65 E-5	3.30 E-6	1.7 E-8	2.0 E-9
New Plant (AP-1000) single unit on site	2.86 E-7	2.32 E-9	1.37 E-10	3.42 E-12
Current generation single unit and single unit New Plant (AP 1000)	3.68 E-5	3.31 E-6	1.71 E-8	2.0 E-9
Two operational units and two New Plants (AP 1000)	7.36 E-5	6.62 E-6	3.42 E-8	4.0 E-9

Continued

Table 10 Continued

Site Configuration	Site CDF	Site LERF	Site Risk (Early Fatalities)	Site Risk (Latent Cancers)
Two operational units and two New Plants (AP 1000) with modified CDFs	7.55 E-5	6.74 E-6	3.68 E-8	4.1 E-9
One High Temperature Gas Reactor (150 MWe)	8.73 E-6	8.73 E-6	0.0	2.04 E-10
Two High Temperature Gas Reactors (300 MWe)	1.75 E-5	1.75 E-5	0.0	4.08 E-10
Eight High Temperature Gas Reactors (1200 MWe)	6.98 E-5	6.98 E-5	0.0	1.63 E-9
Two operational units and Eight High Temperature Gas Reactors	1.43 E-4	7.64 E-5	3.4 E-8	5.63 E-9

The next concern would be how to operationally split the CDF and LERF site objectives between the particular plants on a site. As discussed in Chapter 5 the 1.0 x 10-4 CDF per site year goal can be allocated to the plants on the site. For example is we use the two operational units and two new plants (AP 1000) case in table 10 we have the power adjusted site CDF = 7.36 x 10-5 which meets the CDF objective. For each plant we have

AP-1000 #1 = 2.86 x 10^{-7} < 4.5 x 10^{-6} (per plant year)
AP-1000 #2 = 2.86 x 10^{-7} < 4.5 x 10^{-6} (per plant year)
Current Reactor #1 = 3.65 x 10^{-5} < 4.5 x 10^{-5} (per plant year)
Current Reactor #2 = 3.65 x 10^{-5} < 4.5 x 10^{-5} (per plant year).

As can be seen, the modified subsidiary goals and QHOs can be met for current and new plants and are practical for implementation. There are only two areas where there may be an issue with implementing these new site subsidiary objectives. The first is for sites with two or more current generation plants. Although for the plants shown above the site CDF and LERF objectives were met, there are a few plants that have higher base CDFs and LERFs that would likely not meet the modified subsidiary objectives. This would be particularly true for the three unit sites. One possible method of dealing with this issue would be to permit these plants to use the QHOs instead of the subsidiary objectives, because of the extensive margin in between them. However this would require the plants to complete a level 3 PRA to support that the margin is available for their particular situation.

The second issue is with the HTGRs not meeting the LERF site objectives. HTGRs tend to have much smaller inventories for radionuclide to release and because of the dynamics associated with their fuel failure, tend to have source terms that are much smaller and take longer to release. This coupled with the fact that in general HTGRs and the older GA HTGR used in this analysis tend to have much weaker containments (or confinements) leads to the conclusion that the use of LERF as a safety metric makes less sense for these reactors than for current generation plants. As discussed above, the logical method for dealing with this issue is to redefine the LERF to only releases that can cause prompt fatalities.

6.3 Evaluation of the New QHO

To evaluate potential for more than one plant on a site the new comparative technology QHO will need to use the prompt fatalities, latent cancer deaths, and the offsite consequences for the multi-plant analysis completed in Section 6.2. The other aspects of the structured value problem will be taken from the analysis completed in Chapter 4. With this completed we can evaluate the practicality of this new QHO.

In addition to the evaluation of the prompt fatalities, the latent cancer fatalities, and the land and water contamination that needs to go into the structured value problem for nuclear power plants there also needs to be some way for accounting for the back end of the fuel cycle. As part of analyzing the hazards of nuclear power operation, one of the potential pathways for releasing radionuclei is the failure of used fuel while in storage. While the current political situation makes the final determination of future of the back end of the fuel cycle in the U.S. unsure at least an attempt needs to be made to look at these potential risks. Currently, used fuel (also known as spent fuel) is stored in the spent fuel pools at all U.S. nuclear power plants for a number of years after having been removed from the core. When the fuel has decayed sufficiently to permit its transfer, it is now moved to dry cast storage facilities. Up until recently the plan was to move the used fuel to a geological repository for permanent disposal. In that case its risks would have to be accounted for during four phases: while in the spent fuel pools, while in dry cast storage (including movement from the spent fuel pool to the dry cast), while in transport to the geological repository and while in the geological repository. While in the spent fuel pool, the risk is generally included in the risk of the reactor facility and would be included in the general PRA of the nuclear reactor. Since the exact solution for the transportation and ultimate permanent storage of used fuel has not been determined, estimates for these risks will need to be used. For the phase in which the used fuel is in dry storage, a PRA has been completed [63] and this information will be used to support this part of the analysis. No prompt fatalities are expected and the individual probability of a latent cancer fatality of is 1.8×10^{-12} during the first year of service, and 3.2×10^{-14} per year during subsequent years of storage.

Using this information as an added factor for the structured value of a nuclear plant, we can now evaluate several of the site configurations shown in Table 10. Below in Figures 12, 13, 14 and 15 are the structured value analysis for two operational units and two new plants, and the two operational units and eight HTGRs configurations for both with and without risk aversion considered.

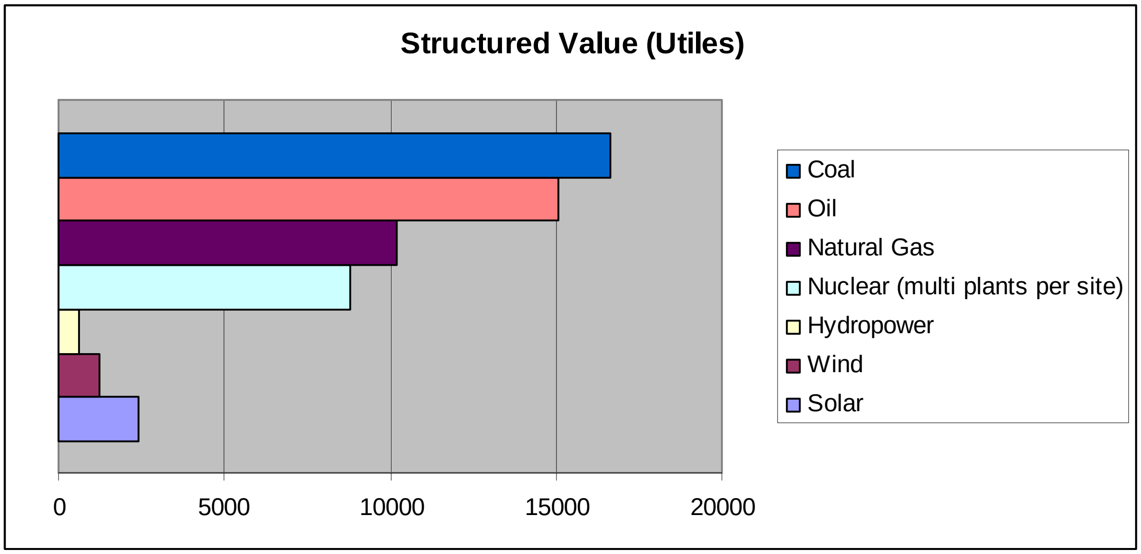

Figure 12: Total Risk for Alternative Generation Technologies (two operational units and two new plants, adjusted for risk aversion)

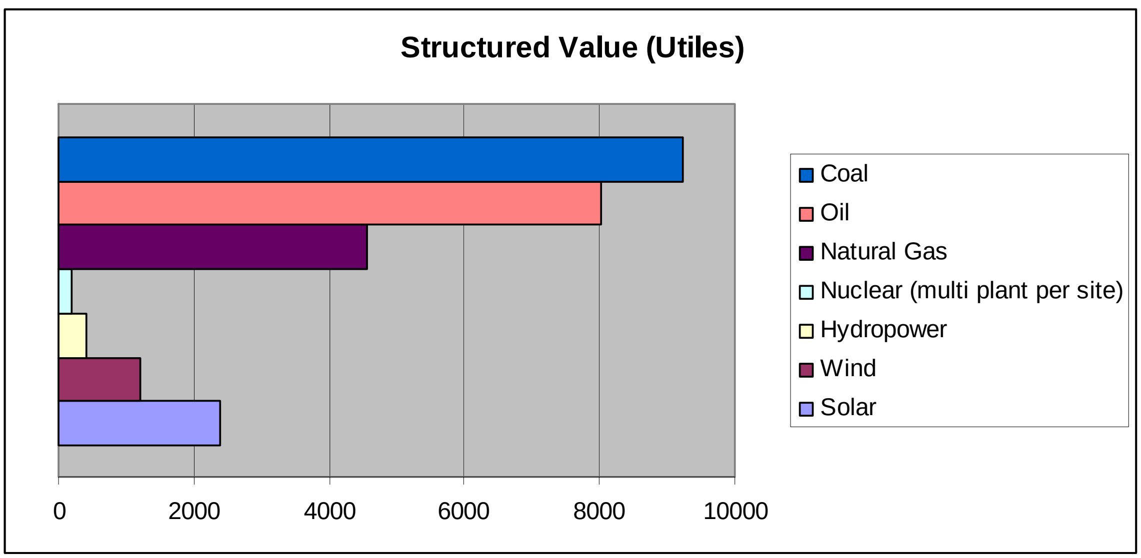

Figure 13: Total Risk for Alternative Generation Technologies (two operational units and two new plants, without risk aversion)

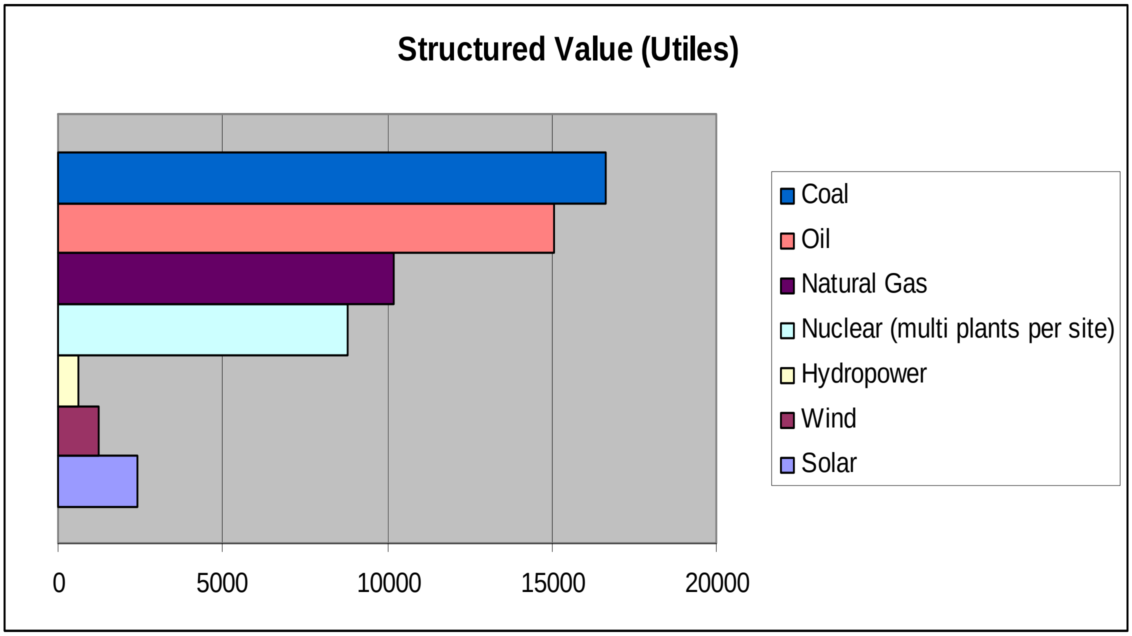

Figure 14: Total Risk for Alternative Generation Technologies (two operational units and eight HTGRs, adjusted for risk aversion)

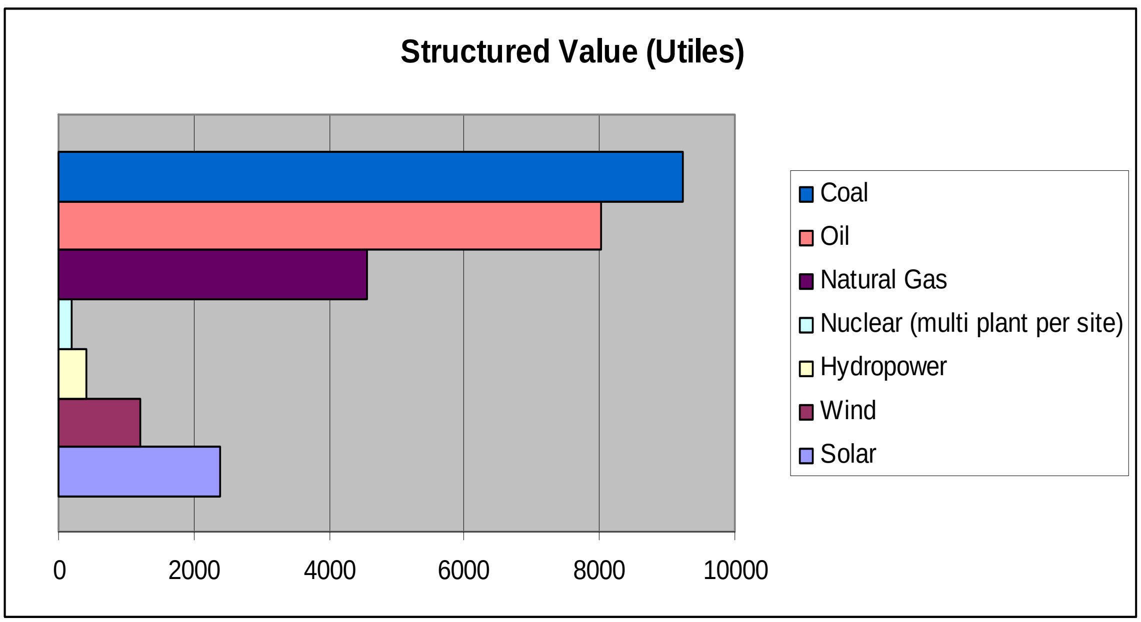

Figure 15: Total Risk for Alternative Generation Technologies (two operational units and eight HTGRs, without risk aversion)

As can be seen, the addition of multiple plants on a site does not significantly affect this QHO. The use of the proposed structured value risk of approximately 10,000 utiles for the case where risk aversion is applied and 4,500 utiles, where risk aversion is not applied can be effectively used.

In this chapter it has been shown that for most cases (with the exception of some current generation sites with high base CDFs) the modifications to the current subsidiary objectives for power level and combined site risk provide a workable method for updating the safety goals to better account for these issues as well as for assigning the appropriately lower risk thresholds to new plants. Additionally it has demonstrated that for representative site configurations the new QHO could also be practically used.

CHAPTER 7: UNCERTAINTY AND SENSITIVITY ANALYSIS

7.1 Uncertainty in Setting the Safety Goals

In the setting of the safety goals there are a number of important assumptions inherent in the setting of the thresholds for the QHOs and risk surrogates, including the effect of uncertainties in the risk and other negative health effects data. Uncertainty is present in the data on fatalities and life cycle health effect, amount of greenhouse gases (e.g., carbon dioxide) produced as part of electric production, and accident frequency and consequences. Additionally there are a number of modeling assumptionsincluding the role of risk perception, the use of thermal power as a surrogate for plant source terms, and the effects of multi-plant failures on the subsidiary goals that could significantly affect this analysis. Uncertainty and sensitivity analysis will be discussed in this chapter.

7.2 Uncertainty in Application of the Subsidiary Objectives

In application of the subsidiary objectives there is significant uncertainty in the calculation of the CDF and LERF as well as the consequence analysis. To support the use of the new subsidiary objectives and QHOs the uncertainty of these calculations should to be considered. The uncertainty in CDF for the plants that were used in this analysis is substantial. The ranges for the 5% to 95% confidence for the Surry CDF and LERF are more than two orders of magnitude (6.8×10^{-6} to 1.3×10^{-4} per yr for CDF). These values will push the CDF and LERF for any of the site configurations well above the new site thresholds if the 95% confidence limit is used. However this is the same for the current subsidiary objectives and their use in the current regulatory structure. It is proposed here that this should not be changed in the application of the new objectives. However, where plant CDF and LERF are suspect there needs to be a close review of the uncertainties to determine if the thresholds might be in jeopardy. As discussed in Chapter 6, the extremely low values of CDF and LERF for the AP-1000 plant should be reviewed to assure that the values obtained at the stage of design certification are not overly optimistic relative to what can be expected when the plant is operational.

Because of the significant margin that exists between the QHOs and the subsidiary goals, more than a factor of 50 based on the NUREG-1150 analysis [34], for most site configurations this will not be an issue.

The assumptions associated with the use of thermal power as a surrogate for plant source terms and the effects of multi-plant failures on the subsidiary goals will also introduce uncertainty in the analysis of the subsidiary objectives. As discussed in Chapters 4 and 6, for light water reactors these uncertainties are expected to be small compared to the uncertainties in the CDF and LERF numbers themselves (5-10% compared to two orders of magnitude). Although it is recommended that, for plants that might have significant dependencies between components, a detailed analysis should be performed, as was done in the IPE [32] program, it is unlikely that the effect on site-wide subsidiary goals will be significant.

7.3 Uncertainty in Application of the New QHO

The new QHO on risk comparison with other competing technologies is based on a number of analysis assumptions as well as data that may be suspect. Reviewing a number of these assumptions to determine if the data used will significantly affect the acceptance criteria (approximately 10,000 utile or 4,500 utile) will provide some insight to how sensitive the method is to uncertainty in the data. The first of these issues is how well the accident, normal operation, and life cycle health and safety data are known. For nuclear power these data were developed using well established methods that have been extensively peer-reviewed. For non-nuclear technologies the data have been collected from previously published research and databases. In some cases there is a relatively well established method for collecting and analyzing the data (health risk of coal and oil production of electricity, maintained by the EPA and others). However, for newer technologies data are hard to come by and methods of analysis are new and unproven [39]. Some studies indicate that there is a large amount of variability in these numbers [39]. Table 11 provides an indication of the size of this variability. Although there has not been an in-depth study of values for hydro, wind and solar, the general lack of data and the variation reported for other technologies would indicate that the uncertainty will be at least as large as reported [39] and summarized here for coal, oil, natural gas and nuclear power. The uncertainty in the other factors in the data associated with the competing technologies, including land and water usage, land and water contamination and greenhouse gas production are relatively small compared to these values (on the order of one order of magnitude). To demonstrate the potential effects of these uncertainties on decision-making Figure 16 provides an illustration of how using the extreme values of this variation could affect the analysis. The upper bar shows the threshold values for comparison (developed using natural gas). The red bar provides the range of values using the extremes of the uncertainty values discussed. As can be seen, although decision can be made

Table 11: Reported Uncertainty in Early Fatality, Normal Operation and Lifecycle Health Effects Data

	Early Fatality (Fatalities/GWe-year)	Normal operation and lifecycle health effects (Fatalities/GWe-year)
Coal	1.1 E-1 to 2.0	1.0 to 1.5 E+2
Oil	1.0 E-3 to 1.0 E-1	5.0 E-1 to 1.0E+2
Natural Gas	8.0 E-4 to 3.0 E-1	6.0 E-3 to 2.0 E-1
Nuclear	4 E-5 to 1.5 E-1	7.0 E-3 to 3.0

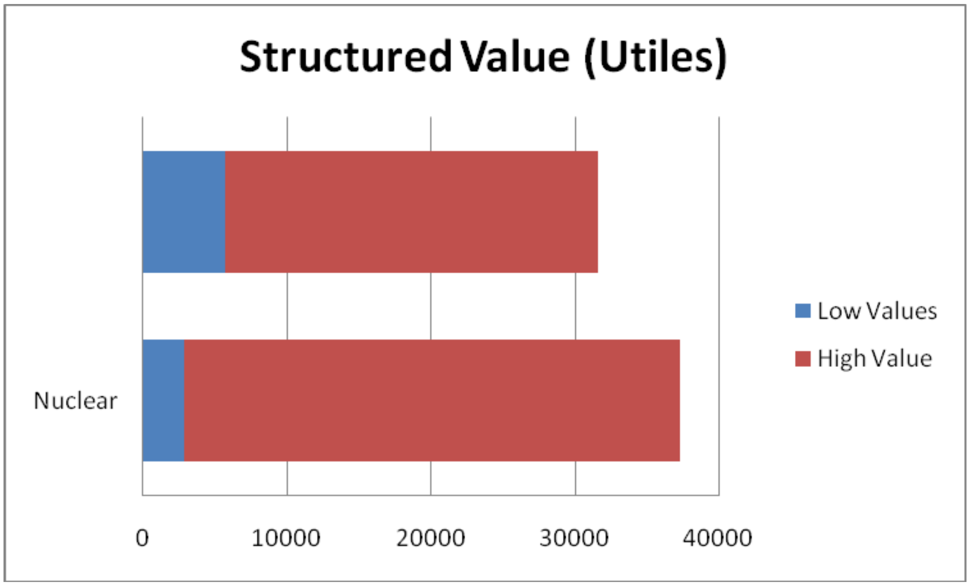

Figure 16: Comparison of Uncertainty for Natural Gas and Nuclear Total Risk for (adjusted for risk aversion)

using mean values, it is important to understand that the uncertainly can significantly affect the decision. Because the amount of data on health and safety consequences of some of the non-nuclear technologies (wind, solar and hydro) is so limited, it makes putting numerical uncertainty bands on the results impractical.

7.4 Sensitivity Studies

In addition to the uncertainty analysis that has been done in the above section it is important to look at some of the analysis methods and assumptions that have gone into the analysis part of this dissertation. As pointed out in Chapter 4 the two most important assumptions have to do with the use of the risk aversion calculations and the value assumed in the greenhouse gas structured value. In Figure 17 we see that a higher value for the assumed values for greenhouse gas will provide a significantly higher value for natural gas (the benchmark threshold value) while not significantly raising the value for nuclear. The value used here for the "higher value" of greenhouse gases is the number proposed by several European countries to reduce greenhouse production (~$100 per metric ton) as opposed to the current value on the market (~$25 per metric ton).

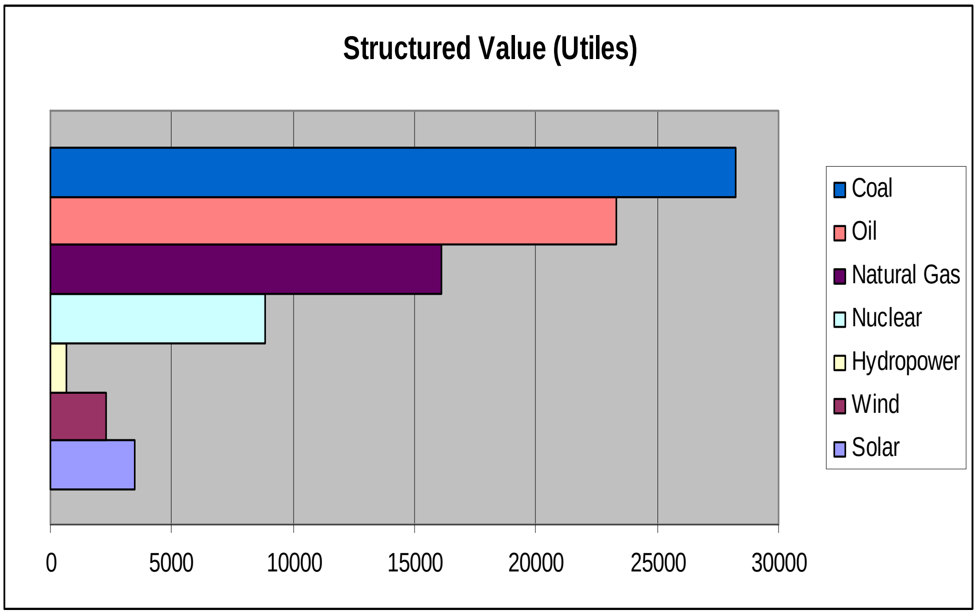

Figure 17: Total risk for Alternative Generation Technologies (with higher greenhouse gas values, adjusted for risk aversion)

Also of importance are the values that are placed on the risk aversion calculations. As shown in Chapter 4 and 6, this value has significant effect, particularly with respect to nuclear power. Figures 18 and 19 show the total risk for all technologies without the risk aversion correction factors. Figure 19 provides this information in the more traditional (for risk analysis) log format. This format provides a better perspective in the comparison of the risk of alternatives when there are order of magnitude uncertainties in the values.

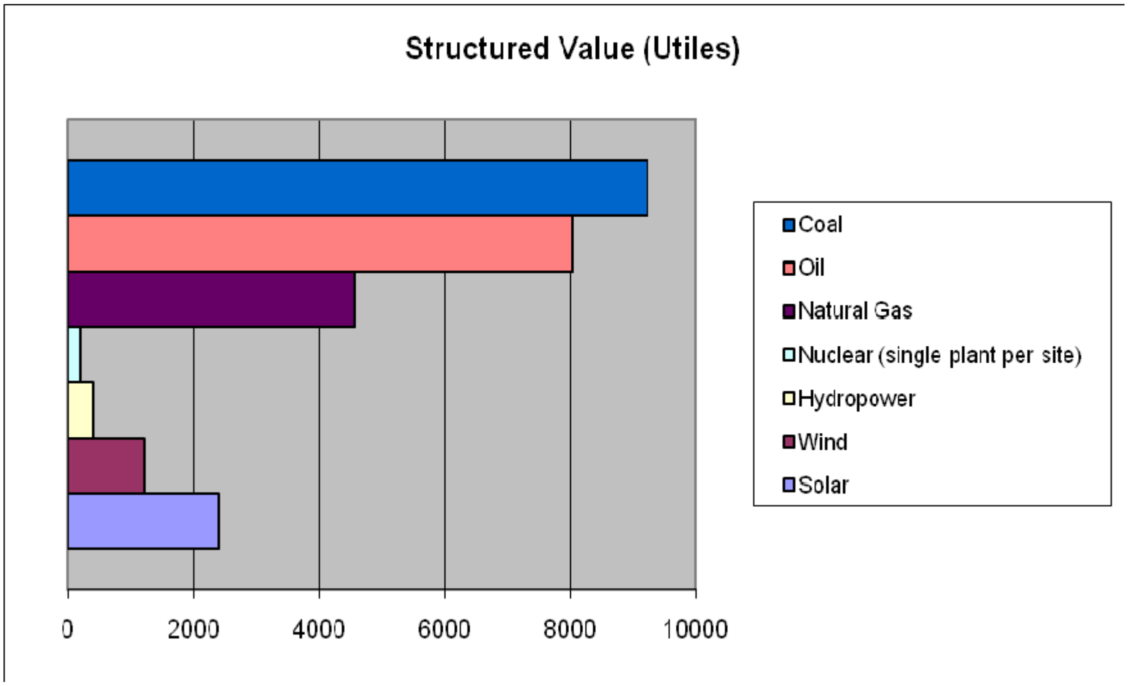

Figure 18: Total Risk from All Factors for Alternative Generation Technologies (without risk aversion)

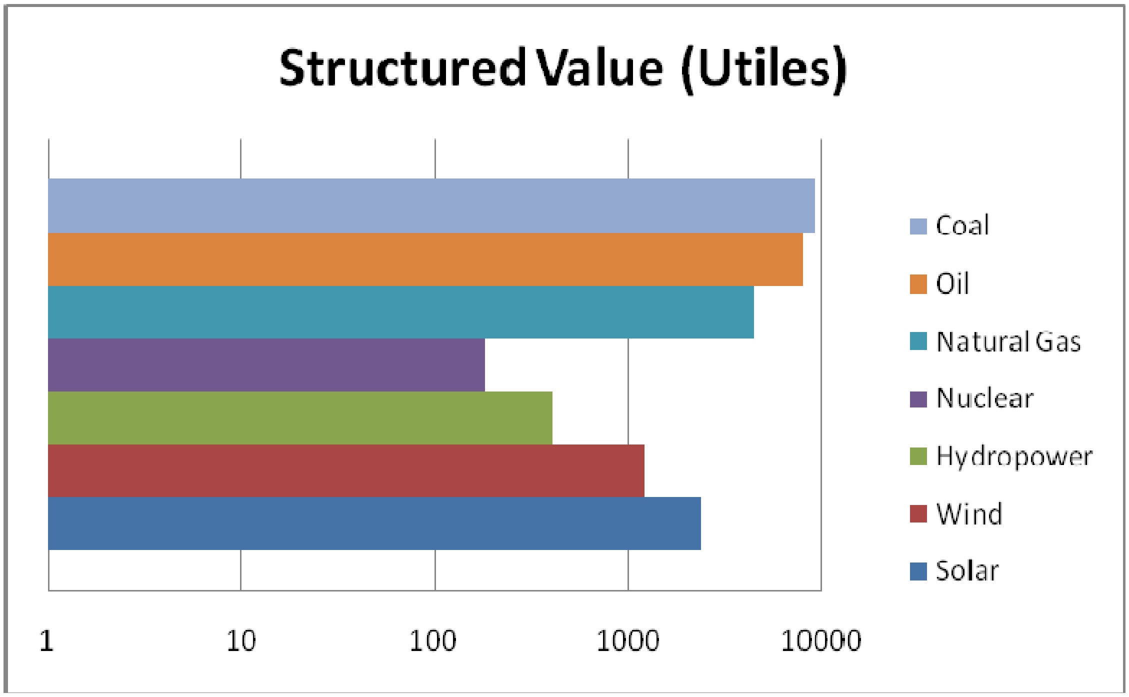

Figure 19: Total Risk from All Factors for Alternative Generation Technologies (without risk aversion)

Regardless of the preferred presentation format, the objective of having a metric for the comparison of competing technologies for the generation of electricity has been established. The use of the new QHO, with the appropriate cautions, can be accomplished. The effect of a much higher risk aversion for nuclear power than natural gas (by a factor of more than 3) compared to wind provides natural gas with an advantage when taking risk aversion into account. However, nuclear has less total negative effect in all cases than natural gas. With respect to wind, if a slightly lower risk aversion (3.9 times lower) would be applied to nuclear power then is would on par with wind even with remaining absolute risk aversion of more than 30 compared to wind. Table 12 provides some additional insight associated with this comparison. This demonstrates that if risk aversion is used in this analysis, more research is needed into better determining the appropriate risk aversion to use for each technology, as fairly small changes can affect the decision.

Table 12: Effects of Risk Aversion

	Structured Value with No Risk Aversion	Structured Value with Full Risk Aversion	Structured Value with Reduced Risk Aversion for Nuclear by factor of 2	Structured Value with Reduced Risk Aversion for Nuclear by factor of 3.8
Natural Gas	4553	20203	20203	20203
Nuclear	182	8845	4431	1217
Hydro	403	597	597	597
Wind	1217	1217	1217	1217
Solar	2399	2399	2399	2399

CHAPTER 8: CONCLUSIONS AND FUTURE RESEARCH NEEDS

This dissertation has examined potential approaches to updating the safety goals that include the establishment of new quantitative health objectives associated with the comparative risk of generating electricity by viable competing technologies, and modifications of the subsidiary objectives to account for multi-plant reactor sites. Issues associated with the use of safety goals in both initial licensing and operational decision making have been examined.

This research developed a new quantitative health objective that uses a comparable benefit risk metric based on the life-cycle risk of the construction, operation and decommissioning of comparable non-nuclear electric generation facilities, as well as the risks associated with mining and transportation. This dissertation also evaluated the effects of using various methods for aggregating site risk as a safety metric, as opposed to using single plant safety goals.

Additionally, a number of important assumptions inherent in the current safety goals, including the effect of other potential negative societal impacts such as the generation of greenhouse gases (e.g., carbon dioxide) have on the risk of electric power production and their effects on the setting of safety goals have been explored. Finally the role risk perception can play in establishing safety goals, has been explored. To complete this evaluation a new method to analytically compare alternative technologies of generating electricity was developed, including development of a new way to evaluate risk perception, and a new method was developed for evaluating the risk at multiple units on a single site.

To test these modifications to the safety goals a number of possible reactor designs and configurations were evaluated using these new proposed safety goals to determine the goals' usefulness and utility. The results of these analyses showed that the modifications provide measures that more closely evaluate the potential risk to the public from the operation of nuclear power plants than the current safety goals, while still providing a straight forward process for assessment of reactor design and operation.

Although the continued use of CDF and LERF as the primary tools for monitoring and regulating the risk of nuclear power plants will continue to impose some limits to the level of accuracy associated with the evaluation of offsite consequences, the modification proposed to the subsidiary objectives based on thermal power and number of plants on a site (site based subsidiary objectives) will better approximate the potential net off-site consequences to the public.

As previously discussed in Chapter 2 and 4 above one of the limitations of the current subsidiary objectives is the use of LERF. LERF is poorly defined and does not capture the entire risk to the offsite public nor does it appropriately account for all reactor types. The revision of LERF is beyond the scope of this research, primarily because high quality level two PRA information including realistic release estimates to the environment is not generally available for current reactor designs. However an approach to improving the treatment of this subsidsidary objective is discussed in Chapter 4.

The proposed subsidiary goals that account for power and the number of reactors per site provide more stringent CDF and LERF criteria for plants already operating at sites that could present a challenge to satisfy. One possible means by which to mitigate this challenge is to allow the nuclear power plant applicant to perform a full site level 3 PRA and base the evaluation on QHOs for which the margin is typically greater than for CDF and LERF. Alternatively, the CDF and LERF subsidiary objectives could be relaxed. This would only be appropriate if the site CDF and LERF uncertainty analyses could support this relaxation, and a detailed review of the margin between the new site CDF and LERF and the QHOs could be provided.

The primary uncertainties associated with the new power adjusted site CDF and LERF subsidiary objectives are associated with calculation of the plant CDF and LERF numbers, rather than the assumptions associated with the calculation of the site criteria. Nevertheless, the additional uncertainties associated with the use of thermal power as a surrogate for the source term in the power adjustment and the assumption of independence in aggregating the site CDF and LERF were examined. Although the evaluation in this dissertation indicates that these contributions to uncertainty are relatively small, this should be studied more fully for other plant configurations involving higher dependencies among units than is currently typical for a plant site.

The new QHO associated with the comparison to other competing technologies for the generation of electricity can be a powerful tool in both the regulation and communication of nuclear power risks. Although its use was discussed only briefly in this dissertation, it's potential for expanded uses in the analysis of advanced reactor regulation should be explored. The new QHO was shown to provide an effective way to compare the effects of negative societal risk for different competing ways to generate electricity. For the cases studied nuclear power was shown to be the lowest risk option over all other current base load alternatives. The new QHO also demonstrates that by keeping nuclear power risk at or below the other QHOs the NRC and the nuclear industry have achieved the intent of the qualitative safety goals and that the one tenth of one percent quantitative thresholds are appropriate.

By developing a new method for incorporating the concept of risk aversion into the new QHO, this dissertation was able to include the effects of risk perception into the analysis of comparative risks for generation of electricity. It was found that this factor can make the comparative risk of nuclear generation either higher or lower than wind or solar. Although the risk of nuclear generation will in all cases be less of a public risk than current base load technology such as coal, the effect of risk perception is a strong factor in comparison with other non fossil fuel alternatives.

Additionally the amount of weight that the negative effect of greenhouse gas generation has on total risk will significantly affect the results of the analysis. However, the value of risk associated with the generation of greenhouse gas does not affect the decision. Nuclear is still preferable to coal and natural gas and to solar and wind even if high values of risk aversion are assumed. The insight gained is that although uncertainties are high, nuclear appears to be the best option (lowest structured value) when risk aversion is low, and in all cases better than coal or natural gas.

The new QHO with the ability to evaluate risk perception provides the NRC and the nuclear industry a method for evaluating the appropriateness of the QHOs and subsidiary objectives. Although the "true" risk is the only proper way to regulate nuclear power operation, this risk perception will permit the decision makers to make a more informed determination of the appropriateness of the risk goals, compared to other technologies.

Potential areas of future research that would significantly improve the industry's ability to evaluate plant performance within the context of the safety goals include:

- Development of more data and better analysis capabilities, particularly for advanced reactors, and lifecycle health risks. The current uncertainties in this information make decision-making a challenge. This is particularly true for data for non-nuclear lifecycle health risks, which is so sparse that the evaluation of uncertainty is difficult.
- Development of full level 3 PRAs for plants including multi-plant accidents, which do not have substantial margin in satisfying site subsidiary goals.
- Detailed analysis of potential multi-plant interactions to determine if the validity of the assumption that the "cross-term" in Equation 4.5 is small is valid.
- Additional analysis into the release frequencies for high temperature gas reactors. Currently available analysis (in the open literature) does not provide sufficient information to adequately assess the applicability of the safety goals. This information is likely to become available in support of the Next Generation Nuclear Power Plant development being supported by the Department of Energy.

- Additional research in the development of more appropriate technology neutral subsidiary risk metrics. Metrics such as an appropriately defined LRF would be particularly useful for non-light water reactor such as high temperature gas reactors or liquid metal reactors.
- Additional use of the proposed new safety goals, particularly the new QHO, in regulatory reviews, to gain experience in their application. The most effective way to do this is to develop a research pilot program where the new safety goals would be used in parallel with the current goals to assess the advantages and limitations of the proposed modified subsidiary objectives and the new QHO.

Additionally, what is needed is to examine the impact on regulatory decisions of the changes proposed in this study for a variety of applications to determine if they can be implemented as discussed in Chapter 5. Based on the review in this dissertation, it appears that they can and they will provide an improved method for assessing the safety of nuclear power plants, and demonstrating to the public that nuclear power plants are "safe enough" to continue to provide a significant part of the U.S. electric power supply.

REFERENCE

1. U.S. Nuclear Regulatory Commission Policy Statement, "Safety Goals for the Operations at Nuclear Power Plants," Federal Register, Vol. 51, (51 FR 30028, August 4, 1986 (ADAMS ACCESSION NUMBER ML051580410).
2. U.S. Nuclear Regulatory Commission, *Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, WASH-1400, Washington D.C, October 1975.
3. U.S. Nuclear Regulatory Commission, "Modifications to the Safety Goal Policy Statement", SECY-99-191, July 22, 1999.
4. Letter dated, September 21, 2005, from Graham B. Wallis, Chairman, Advisory Committee on Reactor Safeguards to Nils J. Diaz, Chairman, U.S. Nuclear Regulatory Commission, "Report on Two Policy Issues Related to New Plant Licensing."
5. U.S. Nuclear Regulatory Commission Policy Statement, "Regulation of Advance Nuclear Power Plants," Federal Register, Vol. 59, (59 FR 35461), July 12, 1994.
6. A.M. Weinberg, "Science and Trans-Science," *Minerva,* Vol. 10, pp 209-222, 1972
7. U.S. Nuclear Regulatory Commission, "Implementation of the Safety Goals", SECY-89-102, December 11, 1989.
8. U.S. Nuclear Regulatory Commission, "Staff Requirements Memorandum - SECY-89-102 – Implementation of the Safety Goals," June 15, 1990.
9. J.G. Kemeny, *Report of the President's Commission on the Accident at Three Mile Island: The Need for Change: The Legacy of TMI*, Washington, D.C. October 1979.
10. U.S. Nuclear Regulatory Commission, *An Approach to Quantitative Safety Goals for Nuclear Power Plants*, NUREG-0739, Washington, D.C., October 1980
11. U.S. Nuclear Regulatory Commission, Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing, NUREG-1860, Washington, D.C., December 20007.
12. U.S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, Revision 1, November 2002.
13. Letter dated August 15, 1996, from T.S. Kress, Chairman, Advisory Committee on Reactor Safeguards to Shirley Ann Jackson, Chairman, U.S. Nuclear Regulatory Commission, "Risk-Informed, Performance-Based Regulation and Related Matters."

14. U.S. Nuclear Regulatory Commission, "Modifications to the Safety Goal Policy Statement", SECY-00-0077, March 30, 2000.
15. U.S. Nuclear Regulatory Commission, "Staff Requirements Memorandum - SECY-00-0077 - Modifications to the Safety Goal Policy Statement," June 27, 2000.
16. U.S. Nuclear Regulatory Commission, "Modified Reactor Safety Goal Policy Statement", SECY-01-0009, January 22, 2001.
17. U.S. Nuclear Regulatory Commission, "Staff Requirements Memorandum - SECY-01-0009 - Modified Safety Goal Policy Statement," April 16, 2001.
18. U.S. Nuclear Regulatory Commission, "Policy Issues Related to New Plant Licensing and Status of the Technology-Neutral Framework for New Plant Licensing", SECY-05-0130, July 21, 2005.
19. U.S. Nuclear Regulatory Commission, "Staff Requirements Memorandum - SECY-05-130 - Policy Issues Related to New Plant Licensing and Status of the Technology-Neutral Framework for New Plant Licensing," September 14, 2005.
20. U.S. Nuclear Regulatory Commission, Final Significance Determination for the Red Finding and Notice of Violation at Indian Point 2 (NRC inspection Report 05000247/2000-010), Washington, D.C. November 20, 2000.
21. M.S. Baram, "Cost-Benefit Analysis: An Inadequate Bases for Health, Safety, and Environmental Regulatory Decisionmaking," *Ecology Low Quarterly,* Vol. 8, pp 473-531, 1980.
22. T.O. Tengs, M.E. Adams, J.S. Pliskin, D.G. Safran, J.E. Siegel, M.C., Weinstein, and J. D. Graham, "Five Hundred Life-Saving Intervention and Their Cost Effectiveness," *Risk Analysis,* Vol. 15(3), pp 369, 1995.
23. B. Black, F. Niehaus and D. Simpson, "How Safe is Too Safe?," *International Institute for Applied Systems Analysis,* WP-79-68, June 1979.
24. Lind, N., "Social and Economic Criteria of Acceptable Risk", *Reliability Engineering and Systems Safety*, Vol. 78 (2002), pages 21-25.
25. W.D. Rowe, *An Anatomy of Risk,* Wiley, New York, N.Y., 1977.
26. C. Starr, "Social Benefit Versus Technological Risk," *Science,* Vol. 165, pp 1232-1238, 1969.
27. B.C. Fischhoff, C. Hohenemser, R.E. Kasperson and R.W. Kates, "Handling Hazards: Can Hazard Management be Improved?," *Environment,* Vol. 20 (7), pp 16-37, 1978.
28. C.A. Adams and C.N. Stone, "Safety and Siting of Nuclear Power Stations in the United Kingdom," *Proceeding of the IAEA Symposium on the Containment and Siting of Nuclear Power Plants,* Vienna, April 1967.
29. G.H. Kinchin, "Design Criteria, Concepts and Features Important to Safety and Licensing," *Proceedings of the International Meeting on Fast Reactor Safety Technology*, Seattle, Washington, 1979.
30. U.S. Atomic Energy Commission, *Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants,* WASH-740, Washington, D.C., 1957.
31. F. Farmer, "Reactor Safety and Siting; A Proposed Risk criterion," Nuclear Safety, 539-548, 1967.

32. Keller, W., and M. Modarres, “A Historical Overview of Probabilistic Risk Assessment Development and Its Use in the Nuclear Power Industry: a Tribute to the Late Professor Norman Carl Rasmussen, *Reliability Engineering and Systems Safety*, Vol. 89, pages 271-285, 2005.
33. U.S. Nuclear Regulatory Commission, H. Lewis, et. al., *Risk Assessment Review Group Report,* NUREG/CR-0400, 1978.
34. U.S. Nuclear Regulator y Commission, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,* NUREG-1150, December 1990.
35. U.S. Nuclear Regulatory Commission, Generic Letter 88-20, “Individual Plant Examinations for Severe Accident Vulnerabilities,” November 23, 1988.
36. U.S. Nuclear Regulatory Commission Policy Statement, “Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities,” August 16, 1995.
37. U.S. Nuclear Regulatory Commission, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Regulatory Guide 1.200, January, 2007.
38. H. Inhaber, *Risk of Energy Production,* AECB-119/Rev-2, Atomic Energy Control Board, Ottawa, Ontario, 1978.
39. D.H. Johnson, T.U. Marston and M.L. Xing, “On a Quantitative Definition of Green,” *Proceedings of the 5th International Conference on Probabilistic Safety Assessment and Management (PSAM-5),* S. Kondo and K. Furuta (Eds.), Osaka, Japan, Nov. 27 – Dec. 1, 2000, 284-288.
40. W. Krewitt, F. Hurley, A. Trukenmuller and R. Friedrich, ‘Health Risks of Energy Systems,” *Risk Analysis,* Vol. 18, No. 4, pp 377-383, 1998.
41. S. Hirschberg, R. Dones and U. Gantner, “Use of External Cost Assessment and Multi-Criteria Decision Analysis for Comparative Evaluation of Options for Electricity Supply”*, Proceedings of the 5th International Conference on Probabilistic Safety Assessment and Management (PSAM-5),* S. Kondo and K. Furuta (Eds.), Osaka, Japan, Nov. 27 – Dec. 1, 2000, 289-296.
42. Nuclear Energy Agency, Organization for Economic Co-Operation and Development, *Risk and Benefits of Nuclear Energy,* 2007.
43. Kennedy, D. “Risks and Risk”, editorial in *Science*, Vol. 309, September, 2005.
44. D. G. MacGregor, P. Solvic and T. Malmfors, “‘How Exposed is Exposed Enough?’ Lay Inferences About Chemical Exposure”, *Risk Analysis,* Vol. 19, pages 649-659, 1999.
45. Bier, V.M., “Challenges to the Acceptance of Probabilistic Risk Analysis”, *Risk Analysis*, Vol. 19, No. 4 (1999), 703-710.
46. M. Modarres, *Risk Analysis in Engineering, Techniques, Tools, and Trends,* CRC Taylor & Francis, Boca Raton, FL, 2006.
47. U.S. Department of Energy, *Carbon Dioxide Emissions from the Generation of Electric Power in the United States,* Washington, D.C., July 2000.
48. K.N. Fleming, “On the Issue of Integrated Risk-A PRA Practitioner Perspective,” *Proceedings of the AMS International Topical Meeting on Probabilistic Safety Analysis,* San Francisco, California, September 11-15, 2005, American Nuclear Society, 2000 (CD-ROM).

49. S. Arndt, "A Simple Method for Assessing Risk from Multiple Reactors on a Site," *Transactions of the American Nuclear Society*, November 2009.
50. U.S. Nuclear Regulatory Commission, *Code Manual for MACCS2,* NUREG/CR-6631, 1998.
51. P. Slovic, "Informing and Education the Public About Risk," *Risk Analysis,* Vol. 6(4), pp 403, 1986.
52. P. VanDoren, *Cato Handbook for Congress: Policy Recommendations for 108th Congress,* Cato Institute Report, 2003.
53. J. Graham, and J.W. Vaupel, "Value of a life: What Difference Does It Make?" *Risk Analysis*, Vol. 1(1), pp 89-95, 1981.
54. U.S. Nuclear Regulatory Commission, *Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission,* NUREG/BR-0058, Rev 4, 2004.
55. W.D. Rowe, *An Anatomy of Risk,* John Wiley & Sons, New York, 1977.
56. P. Slovic, "Perception of Risk," Vol. 236, pp 280, 1987.
57. V.T. Covello, D.B. McCallum, and M.T. Pavlova (Eds.), *Effective Risk Communication: The Role and Responsibility of Government and Nongovernment Organizations*, Plenum Press, New York, 1989.
58. C. Starr, "Social Benefit versus Technical Risk," *Science*, Vol. 165, pp 1232-1238, 1969.
59. U.S. Nuclear Regulatory Commission, "Proposed Agency-Wide Implementation Plan for Probabilistic Risk Assessment (PRA)", SECY-94-219, 1994.
60. Office of Management and Budget, "Regulatory Analysis," Circular No. A-4, September 17, 2003.
61. U.S. Nuclear Regulatory Commission, *Systems Analysis Programs for Hands-on Reliability Evaluations (SAPHIRE),* NUREG/CR-6952, 2008.
62. GA Technologies, *Probabilistic Risk Assessment of the Modular HTGR Plant,* HTGR-86-011, Revision 1, June 1986.
63. U.S. Nuclear Regulatory Commission, *A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant),* NUREG-1864, 2007.