The Use of Frequency-Consequence Curves in Future Reactor Licensing

by

Laurène Debesse

Diplôme d'ingénieur
Ecole Polytechnique, France, 2005

Submitted to the Engineering Systems Division and the Department of Nuclear Science and Engineering in Partial Fulfillment of the Requirements for the Degrees of

Master of Science in Technology and Policy and
Master of Science in Nuclear Science and Engineering
at the
Massachusetts Institute of Technology

February 2007

Signature of Author____
Technology and Policy Program, Engineering Systems Division
Department of Nuclear Science and Engineering
January 5th, 2007

Certified by____
George E. Apostolakis
Professor of Nuclear Science and Engineering and of Engineering Systems
Thesis Supervisor

Certified by____
Michael J. Driscoll
Professor Emeritus of Nuclear Science and Engineering
Thesis Reader

Accepted by____
Dava J. Newman
Professor of Aeronautics and Astronautics and Engineering Systems
Director, Technology and Policy Program

Accepted by____
Jeffrey A. Coderre
Professor of Nuclear Science and Engineering
Chairman, Department Committee on Graduate Students

[This page is left intentionally blank]

The Use of Frequency-Consequence Curves in Future Reactor Licensing

by

Laurène Debesse

**Submitted to the Engineering Systems Division and the Department of Nuclear Science and Engineering on January 5th, 2007
in Partial Fulfillment of the Requirements for the Degrees of
Master of Science in Technology and Policy
and
Master of Science in Nuclear Science and Engineering**

Abstract

The licensing of nuclear power plants has focused until now on Light Water Reactors and has not incorporated systematically insights and benefits from Probabilistic Risk Assessment (PRA). With the goal of making the licensing process more efficient, predictable and stable for advanced reactors, the U.S. Nuclear Regulatory Commission (USNRC) has recently drafted a risk-informed and technology-neutral framework for new plant licensing. The Commission expects that advanced nuclear power plants will show enhanced margins of safety, and that advanced reactor designs will comply with the Commission's Safety Goal Policy Statement. In order to meet these expectations, PRA tools are currently being considered; among them are frequency-consequence (F-C) curves, which plot the frequency of having C or more consequences (fatalities, injuries, dollars, dose...) against the consequences C. The present research analyzes the role and the usefulness of such curves in risk-informing the licensing process in the U.S., and shows that their use allows the implementation of both structuralist and rationalist Defense-In-Depth. The second part of this work concentrates on F-C curves as a mean to assess and limit societal risk. Such tools would improve the safety of current plants by allowing the regulator to focus its attention on the plants that pose the highest societal risks in events such as power uprates.

Thesis Supervisor: George E. Apostolakis, Professor of Nuclear Science and Engineering and of Engineering Systems

Acknowledgements

I would like to gratefully acknowledge the enthusiastic supervision of Professor Apostolakis during this work. I also wish to thank my thesis reader Professor Driscoll, for his help and his very pertinent comments. Finally, I would like to thank the Fulbright Commission, the MIT Nuclear Science and Engineering Department, the Department of Energy and the Nuclear Regulatory Commission for supporting me during my stay at MIT and for giving me the opportunity to participate to extremely challenging and interesting projects.

Table of contents

Acknowledgements ... 4

Table of contents ... 5

Table of figures ... 9

Table of tables ... 11

List of acronyms ... 12

Part I. Introduction ... 14

Part II. Overview of the licensing of nuclear power plants in the United States ... 16

II.A. The Atomic Energy Act and the Energy Reorganization Act ... 16

II.B. Current licensing process ... 16

Part III. Safety philosophy of nuclear power plants ... 19

III.A. Defense-In-Depth ... 19

III.A.1. Definition ... 19

III.A.2. Rationalist and Structuralist Defense-In-Depth ... 20

III.B. Safety Margins ... 20

III.B.1. Epistemic and aleatory uncertainties ... 20

III.B.2. Quantification of safety margins ... 22

III.C. The trend towards risk-informing regulations ... 24

III.C.1. Defining risk quantitatively ... 24

III.C.2. Quantification of risk in the regulations ... 26

III.D. Justifications for a new licensing approach ... 27

III.E. Conclusion ... 28

Part IV. Use of frequency-consequence curves in new reactor licensing ... 30

IV.A. Expected level of safety for future plants ... 30

IV.B. Definition of frequency-consequence curves in the framework ... 31

IV.C. Use of the frequency-consequence curve to implement USNRC's high-level safety expectations ... 32

IV.C.1. Each event sequence must lie individually below the F-C curve ... 32

IV.C.2. The integrated risk is not assessed with the frequency-consequence curve ... 33

IV.D. Use of frequency-consequence curves to identify and select Licensing Basis Events ... 34

IV.D.1. Design-Basis Accidents ... 34
IV.D.1.a. The General Design Criteria ... 35
IV.D.1.b. Design-basis accidents as a tool to show compliance with licensing requirements ... 36
IV.D.1.c. The "Maximum Credible Accident" concept ... 37
IV.D.1.d. The difficulty of dealing with incredible accidents ... 38
IV.D.1.e. Licensing Basis Events should replace Design-Basis Accidents ... 39
IV.D.2. Probabilistic selection of Licensing Basis Events ... 40
IV.D.2.a. Steps for Licensing Basis Event selection ... 40
IV.D.2.b. Criteria for selected Licensing Basis Events ... 42
IV.D.3. Deterministic selection of Licensing Basis Events ... 45
IV.E. Analogy between the current framework draft and Farmer paper ... 46
IV.F. Improvements due to Licensing Basis Events ... 46
IV.G. Conclusion ... 47
Part V. What is societal risk? ... 49
V.A. What is societal risk as opposed to individual risk? ... 49
V.A.1. Individual risk ... 49
V.A.2. Societal risk ... 50
V.A.3. Why isn't putting a limit on individual risk enough? ... 53
V.B. General questions on societal risk ... 54
V.B.1. At what level societal risk should be considered? ... 54
V.B.2. Should risk aversion be included in the societal risk measures? ... 56
V.B.2.a. Definition ... 56
V.B.2.b. Inclusion of risk-aversion into criterion and measure ... 57
V.C. Quantitative risk limits in the United States ... 58
V.C.1. The 1986 Safety Goals ... 58
V.C.2. Surrogate Risk Metrics ... 59
V.D. Sources of societal risk ... 59
V.D.1. Issue of siting ... 59
V.D.2. Plant characteristics ... 61
V.E. Conclusion ... 62

Part VI. Overview of Quantitative tools to measure societal risk 63
VI.A. FN curves 63
VI.A.1. Definition 63
VI.A.2. Different types of FN curves 64
VI.A.3. Use of FN curves in the Netherlands 64
VI.A.4. Dutch regulations and U.S. PRA results 67
VI.A.5. Limits of FN curves 70
VI.B. Other risk assessment measures 70
VI.C. Extended measures of societal risk 72
VI.C.1. Would an extended definition of societal risk be more appropriate? 72
VI.C.1.a. Three Mile Island 72
VI.C.1.b. Chernobyl 73
VI.C.1.c. The number of fatalities does not adequately capture societal risk 74
VI.C.2. Societal risk measures accounting for more than fatalities 74
VI.C.2.a. Swiss proposal of risk measure 74
VI.C.2.b. Swiss criterion 78
VI.D. Conclusion 79
Part VII. Should societal risk criteria be defined in the United States? 81
VII.A. Description of proposal 81
VII.A.1. Overview 81
VII.A.2. Application of proposal 85
VII.A.2.a. Overview of the Generic Environmental Impact Statement for License Renewal of Nuclear Plants 85
VII.A.2.b. Results 86
VII.B. Valuation of life is required in this approach 87
VII.B.1. USNRC policy regarding valuation of life 87
VII.B.2. Sensitivity analysis 88
VII.B.2.a. Latent fatalities dominate 88
VII.B.2.b. Frequency-consequence cost-risk status 89
VII.B.3. Summary of results and implications 93
VII.C. Issued related to the approach 96

VII.C.1. Valuation of injuries ... 96
VII.C.2. Correlation of variables ... 97
VII.C.3. Maturity of computer codes ... 97
VII.C.4. Siting vs Design ... 98
VII.D. Conclusion ... 99
Part VIII. Summary of conclusions ... 101
List of references ... 103
Appendix 1: Overview of NUREG-1437 ... 107
Appendix 2: Valuation of life ... 109

Table of figures

Figure 1: Illustration of epistemic uncertainties 22
Figure 2: Definition of safety margin 22
Figure 3: Design and regulatory margins (USNRC, 2006) 23
Figure 4: Display of uncertainties on F-C curves 25
Figure 5: PRA modeling 26
Figure 6: Three Region Approach to Risk Tolerability 31
Figure 7: Frequency-consequence curve proposed in the USNRC framework 32
Figure 8: Criterion for individual event sequence 33
Figure 9: Parameters of Licensing Basis Events 41
Figure 10: Selection process of Licensing Basis Events 42
Figure 11: Criterion for probabilistically-selected LBE 43
Figure 12: 3-region approach 43
Figure 13: Summary criteria for LBEs 45
Figure 14: Licensing Basis Event Margins 47
Figure 15: Example of FN curve, which displays the probability of having N or more fatalities per year, as a function of N, on a double logarithmic scale 51
Figure 16: Illustration of the difference between individual and societal risk 54
Figure 17: Categories of multiple-fatality risk aversion 56
Figure 18: Siting distances in 10 CFR 100 61
Figure 19: Example of FN curves for different groups of activities in the Netherlands 64
Figure 20: FN criteria in the Netherlands (note that only the upper curve is a criterion) 66
Figure 21: Comparison between Dutch and British risk tolerability criteria (the Canvey Line criterion is risk-neutral, as opposed to the highly risk-averse Dutch criterion.) 67
Figure 22: Example of level-3 PRA results (NUREG-1150) and Dutch criterion 68
Figure 23: Example of level-3 PRA results (NUREG-1150) and risk-neutral criterion 69
Figure 24: Example of level-3 PRA results (NUREG-1150) and risk-averse criterion (slope equal to 1.2) 70
Figure 25: Disaster scale 76
Figure 26: Membership function for total number of fatalities 77

Figure 27: F-C curve proposed by the Swiss Ordinance 79
Figure 28: Complementary Cumulative Density Function for acute fatalities from Plant "X" level-3 PRA 82
Figure 29: Illustrative Complementary Cumulative Density Function F-C risk acceptance criteria (Kress, 2005) 83
Figure 30: Criteria implied by the F-C curve 84
Figure 31: F-C risk-cost status (Kress, 2005) 86
Figure 32: Ratio of the predicted number of early fatalities to the predicted number of latent fatalities 89
Figure 33: Frequency-consequence cost-risk status - value for early and latent fatality is $ 1,000,000 90
Figure 34: Frequency-consequence cost-risk status – Value for early fatality is $2,500,000 and value for latent fatality is $1,000,000 90
Figure 35: Frequency-consequence cost-risk status - Value for early fatality is $2,500,000 and value for latent fatality is $2,500,000 91
Figure 36: Frequency-consequence cost-risk status – Value for early fatality is $4,000,000 and value for latent fatality is $4,000,000 92
Figure 37: Frequency-consequence cost-risk status - Value for early fatality is $12,500,000 and value for latent fatality is $2,500,000 92
Figure 38: Frequency-consequence cost-risk status - Value for early fatality is $20,000,000 and value for latent fatality is $4,000,000 93
Figure 39: Comparison between off-site costs and fatality-related predicted costs (statistical values of early and latent fatalities respectively equal to $12,500,000 and $2,500,000) 94
Figure 40: Assessment of the tolerability of F-C cost risk status using statistical values of early and latent fatalities respectively equal to $12,500,000 and $2,500,000 95
Figure 41: Assessment of the tolerability of F-C cost risk status using statistical values of early and latent fatalities respectively equal to $20,000,000 and $4,000,000 96
Figure 42: High uncertainties for Plant "X" level-3 PRA output 98
Figure 43: Required data depending on the type of frequency-consequence curve 99

Table of tables

Table 1: "Conservative approach" versus "Best estimate approach" (IAEA, 2001) 24

Table 2: Classification of event sequences according to their mean frequency 44

Table 3: Additional deterministic criteria depending on frequency category 45

Table 4: Categories of risks 53

Table 5: Categories defined in Swiss proposal 76

Table 6: Ranking of plants based on their overall societal cost for different values of statistical life 94

Table 7: Valuation of life: Lawsuit of wrongful deaths 110

Table 8: Methods for valuating life 111

List of acronyms

ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
ALARA	As Low As Reasonably Achievable
BWR	Boiling Water Reactor
CCDF	Complementary Cumulative Density Function
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
DBA	Design-Basis Accident
ECCS	Emergency Core Cooling System
EI	Exposure Index
ESP	Early Site Permit
F-C	Frequency-consequence
GEIS	Generic Environmental Impact Statement
GFR	Gas-cooled Fast Reactor
HSE	Health and Safety Executive
IAEA	International Atomic Energy Agency
IE	Initiating Event
LBE	Licensing Basis Event
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LPG	Liquefied Petroleum Gas
LPZ	Low Population Zone
LWR	Light Water Reactor
MACCS	Melcor Accident Consequence Code System
MCA	Maximum Credible Accident
MIT	Massachusetts Institute of Technology
MYR	Middle Year of Relicense
NEA	Nuclear Energy Agency
PBMR	Pebble Bed Modular Reactor

PRA	Probabilistic Risk Assessment
PSAR	Preliminary Safety Analysis Report
QHO	Quantitative Health Objective
RY	Reactor Year
SAR	Safety Analysis Report
TMI	Three Mile Island
USNRC	U.S. Nuclear Regulatory Commission

Part I. Introduction

Nuclear electricity accounts today for approximately 17 percent of worldwide electricity generation. Once regarded as the most promising source of energy, nuclear energy has faced major public opposition heightened by the accidents of Three Mile Island and Chernobyl, which contributed to a slowing down of the whole industry in the United States. Recently, advantages of nuclear power have been given more light and publicity, which fosters the rebirth of nuclear power: among them is the fact that nuclear energy does not contribute to the emission of greenhouse gases. However, fears raised with Three Mile Island and Chernobyl accidents are still vivid. The 2003 Massachusetts Institute of Technology (MIT) study on the Future of Nuclear Power shows that safety is a key discriminating factor to be considered for the growth of nuclear power. In order to address this issue, major changes in the safety approach, for instance the increased use of Probabilistic Risk Assessment (PRA), have been made and contribute to the emergence of a safer fleet of reactors.

All commercial reactors in operation today belong to the Generations II and III. The U.S. Department of Energy's Office of Nuclear Energy, Science and Technology has launched several programs aimed at developing the next generation of nuclear energy systems. Part of the research effort is focused on new reactor concepts, the Generation IV reactors, such as the Gas-Cooled Fast Reactor (GFR), currently designed at MIT. In parallel to the design process currently underway, regulatory authorities are moving forward to define new licensing rules for future plants. Indeed, regulations of nuclear power plants have focused until now on Light Water Reactors only, and have not systematically incorporated insights and benefits from PRA methods. Part II of this work provides an overview of the current licensing process. In Part III, the main concepts of the safety philosophy of nuclear reactors are introduced. Among them is Defense-In-Depth, which will remain a fundamental tenet of the safety approach for advanced reactors.

So, the US Nuclear Regulatory Commission (USNRC) has defined as a goal to risk-inform the regulations and make the licensing process more efficient, predictable, and stable. Indeed, when Title 10 of the Code of Federal Regulations (CFR) Part 50 is used to license a design differing from the Light Water Reactor (LWR) design, the applicability of the regulations must be reviewed, exemptions documented, and additional requirements justified. This case-by-case analysis entails inefficiency. As for the predictability and stability of licensing processes, they pertain to the timing and outcome of the case-by-case review under 10 CFR 50: without a systematic set of rules applicable to all reactors, similar issues might be treated differently and uncertainty on the result of the review arises. To overcome these difficulties, the USNRC has recently drafted a technology-neutral framework for new plant licensing, which should in the long term replace 10 CFR Part 50. An Advance Notice of Proposed Rulemaking was issued by the Commission in May 2006. Similarly, the International Atomic Energy Agency has started giving guidance for developing a set of requirements that would be applicable to any kind of nuclear reactor. An objective of this research work is to analyze the use of specific risk assessment tools known as frequency-consequence (F-C) curves in future reactor licensing. Part IV presents a discussion of frequency-consequence curves in future reactor licensing and shows how such tool allows a risk-informed licensing process.

The question of including societal risk in the regulations has been regularly raised and it is legitimate in the context of the new framework to ask if societal risk should be included in the new licensing approach, and how F-C curves could contribute to societal risk assessment. Part V and VI introduce a different use of frequency-consequence curves as a mean to assess and limit societal risk. Part VII finally discusses the possibility of introducing such societal risk assessment tool in the U.S. regulations.

Part II. Overview of the licensing of nuclear power plants in the United States

The purpose of this part is to present the current licensing process of nuclear power plants. There are two processes for current plants, codified under Code of Federal Regulations (CFR) Title 10 Parts 50 and 52. An alternative licensing process for advanced nuclear plants is currently drafted at the USNRC.

II.A. The Atomic Energy Act and the Energy Reorganization Act

In 1954, Congress amended the 1946 Atomic Energy Act making possible the development of nuclear commercial activities.

The overall policy of the United States towards nuclear energy was defined in Section 1 of the 1954 Atomic energy Act (42 USC 2011), and consisted of two objectives:

> "(a) The development, use, and control of atomic energy shall be directed so as to make the maximum contribution to the general welfare, subject at all times to the paramount objective of making the maximum contribution to the common defense and security;
> (b) The development, use, and control of atomic energy shall be directed so as to promote world peace, improve the general welfare, increase the standard of living, and strengthen free competition in private enterprise."

The Atomic Energy Commission (AEC) was authorized by Section 161(b) of the Act to:

> "establish by rule, regulation or order, such standards and instructions to govern the possession and use of special nuclear material, source material, and byproduct material as the Commission may deem necessary or desirable to promote the common defense and security or to protect or minimize danger to life or property" (42 USC 2201).

The 1974 Energy Reorganization Act established the Nuclear Regulatory Commission (USNRC) to regulate the civilian use of nuclear materials. The Commission, which assumed the regulatory responsibilities of the Atomic Energy Commission, was assigned three regulatory functions: rulemaking, licensing and inspection.

II.B. Current licensing process

Licensing nuclear power plants is under the responsibility of the USNRC. Nuclear power plants currently in operation, all Light Water Reactors (LWRs), have been licensed using a two-step process. They must obtain both a construction permit and an operating license. This process is detailed in 10 CFR Part 50 and briefly summarized below:

- In order to construct or operate a nuclear power plant, the applicant must submit a Safety Analysis Report (SAR), which contains the design information and criteria for the proposed plant, comprehensive data on the proposed site, and also a discussion of hypothetical accident situations and the safety features available for both preventing and mitigating these accidents, should they occur. The application also includes an assessment of the environmental impact of the proposed plant and information for antitrust reviews.
- The USNRC staff reviews the application to determine if the plant design meets all the applicable regulations contained in 10 CFR Parts 10, 50, 73, and 100. This step includes a review of the design of the nuclear plant, the anticipated response of the plant to hypothetical accidents, the emergency plans, and the characteristics of the site. The results of this review are summarized in a Safety Evaluation Report. The Advisory Committee on Reactor Safeguards (ACRS), an independent committee of experts, also reviews the application and submits its results to the Commission.
- If the construction permit is issued, the applicant must then submit a Final Safety Analysis Report to support its application for an operating license.
- The USNRC then prepares a Final Safety Evaluation Report, and the ACRS provides an independent evaluation.

Based on the Atomic Energy Act, commercial power reactor licenses are issued for a 40 year period, with the possibility of renewing the license for 20 years. The first 40-year operating license will expire in 2009. The USNRC has established strict requirements codified in 10 CFR 51 and 10 CFR 54 for license renewal.

In 1989, USNRC established an alternative licensing process codified in 10 CFR 52 in order to improve regulatory efficiency and a greater predictability in the licensing process. An early site permit (ESP) gives a company approval for a plant site before a decision is actually made to build the plant; and resolves site safety, environmental protection and emergency preparedness

issues independently of a particular design. In the design certification process, USNRC examines if the design meets regulatory safety standards. If accepted, the Commission drafts a rule to issue the standard design certification as an appendix to 10 CFR 52.
Finally, a combined license authorizes construction and operation of the facility in a manner similar to a construction permit under the two-step licensing process.

The USNRC Office of Nuclear Regulatory Research is currently taking a step ahead by drafting an alternative to 10 CFR 50, which would be technology neutral, i.e. applicable to all reactor technologies, and risk-informed (USNRC, 2006). Such task calls for new risk assessment tools, such as frequency-consequence curves (F-C curves), for which no previous experience is available. At the same time, the new licensing process must rely on fundamental safety principles such as Defense-In-Depth that have greatly contributed until now to the safety of power plants.

Part III. Safety philosophy of nuclear power plants

The requirements a power plant must fulfill in order to get an operating license have evolved greatly since the licensing of the first plant. They reflect today the two tenets of the safety philosophy: the implementation of Defense-In-Depth and the existence of safety margins, which are an integral part of the Defense-In-Depth concept, but are often discussed separately. Risk-informing the licensing process calls for a greater reliance on risk quantification tools such as F-C curves. In this part, we will describe these two safety principles to later be able to demonstrate how F-C curves maintain both Defense-In-Depth principles and enhanced safety margins.

III.A. Defense-In-Depth

III.A.1. Definition

The concept of Defense-In-Depth has greatly evolved from a "narrow application to the multiple barrier concept to an expansive application as an overall safety strategy" (Sorensen et al, 1999). It is currently interpreted as follows:

- High-level protective strategies are implemented: preventing accident initiators from occurring, terminating or mitigating accidents adequately, preventing degradation or failure of barriers designed to contain radionuclides, and accident management plans to protect the offsite public in case radionuclides penetrate the barriers.
- Multiple physical barriers are required (the "historical" approach).

In a 1999 White Paper on risk-informed and performance-based regulations (USNRC, 1999), the Commission reaffirmed the crucial importance of Defense-In-Depth in its approach to safety:

> "The concept of defense-in-depth has always been and will continue to be a fundamental tenet of regulatory practice in the nuclear field. Risk insights can make the elements of defense-in-depth clearer by quantifying them to the extent practicable. Although the uncertainties associated with the importance of some elements of defense may be substantial, the fact that these elements and uncertainties have been quantified can aid in determining how much defense makes regulatory sense. Decisions on the adequacy of or the necessity for elements of defense should reflect risk insights gained through identification of the individual performance of each defense system in relation to overall performance."

III.A.2. Rationalist and Structuralist Defense-In-Depth

A useful distinction between a "structuralist" model of Defense-In-Depth and a "rationalist" one has been proposed (Sorensen et al, 1999):

- In the *structuralist* approach, "Defense-In-Depth" is embodied in the structure of the regulations and in the design of the facilities built to comply with those regulations. The requirements are derived by constantly asking the question: "what if this barrier fails?" no matter what the probability of failure of the barrier is. Hence, emphasis is put on both accident prevention and accident mitigation. The current safety approach, based on deterministic principles, has relied on the structuralist Defense-In-Depth.
- The *rationalist* model asserts that "defense in depth is the aggregate of provisions made to compensate for uncertainty and incompleteness in our knowledge of accident initiation and progression." This model relies on quantitative acceptance criteria and requires that the system be analyzed using risk assessment methods and the uncertainties be quantified before being managed appropriately.

III.B. Safety Margins

As part of Defense-In-Depth, the main purpose of safety margins is to cope with uncertainties.

III.B.1. Epistemic and aleatory uncertainties

A useful classification of uncertainties into two categories is available in the literature (Apostolakis, 1990, 1993): aleatory uncertainties, which are uncertainties in the model of the world, and epistemic uncertainties, which are uncertainties in the state of knowledge. Such categorization should not be interpreted as if there were in theory two types of probability intended to represent these uncertainties, even if the distinction has useful implications in the modeling of complex systems (Winkler, 1996).

Aleatory uncertainties deal with observable quantities (for instance the time to failure of a component); they come from the fact that events can happen in a random or stochastic manner.

For instance, a pump can fail to start due to a random failure. This type of uncertainty cannot be reduced by further studies, but can be better characterized by additional research. It is usually managed by probabilistic methods.

As opposed to aleatory uncertainties, *epistemic uncertainties* deal with non-observable quantities and arise from our lack of knowledge or lack of scientific understanding. They can be reduced by additional studies and fall into three categories:

- The first category consists of *parameter uncertainty*, which is uncertainty associated with the values of the parameters of the Probabilistic Risk Assessment (PRA) models and the basic data used in safety analysis such as failure rates, ultimate strength, etc. The values of these parameters are not perfectly known.
- The second category of uncertainties, the *model uncertainties*, deals with the uncertainties associated with the data limitations, analytical physical models, and acceptance criteria used in the safety analysis. Experts may formulate different models in order to be as close to reality as possible, even though these models are an approximation of the real phenomena. For instance, model uncertainties arise when modeling human performance or common cause failures such as fires.
- As for the third category, *completeness uncertainty* is the uncertainty associated with factors not accounted for in the safety analysis, such as safety culture, unknown or unanticipated failure mechanisms, etc. It can be considered a scope limitation, whose magnitude is difficult to assess since it reflects an unanalyzed contribution to risk. It has often been referred to as the "unknown unknown".

Let's consider an example to illustrate the differences between these types of uncertainties. A designer might need to assess the values of a certain parameter in a given system or a component on duty, for example the maximum pressure in the containment during a Loss of Coolant Accident (LOCA). Even if the designer had a perfect knowledge of the system, his assessment of the parameter would still be uncertain due to the existence of random phenomena. Moreover, the uncertainties due to its lack of knowledge make the assessment of the parameter even more uncertain: only a probability density function can capture the values of that parameter.

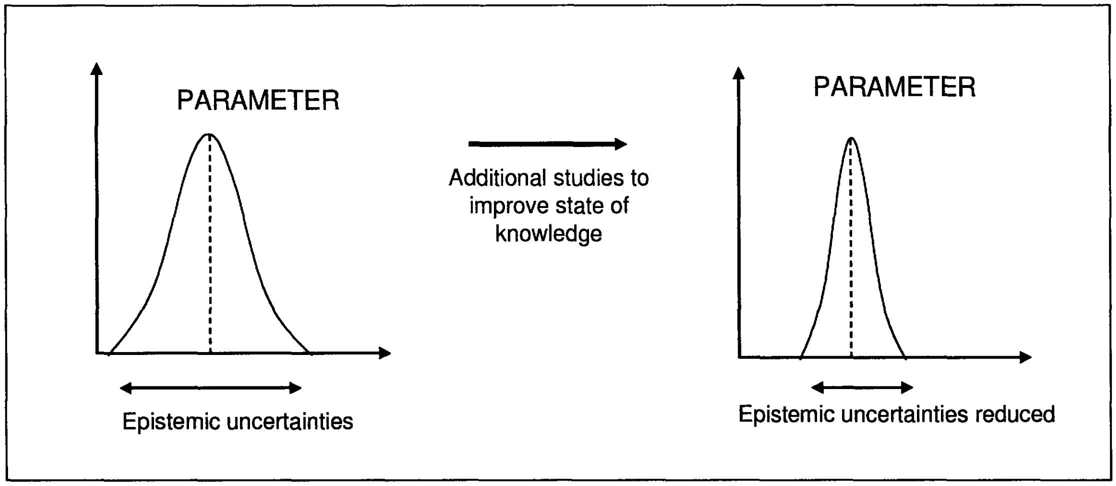

Figure 1: Illustration of epistemic uncertainties

III.B.2. Quantification of safety margins

Safe operation is ensured if safety variables (e.g. peak clad temperature, containment pressure) remain within the capacity limits, defined as the values above or under which the system fails. Safety margin is then defined as the difference between the characteristic value (e.g. the mean value) of the safety variable and the characteristic value of the capacity.

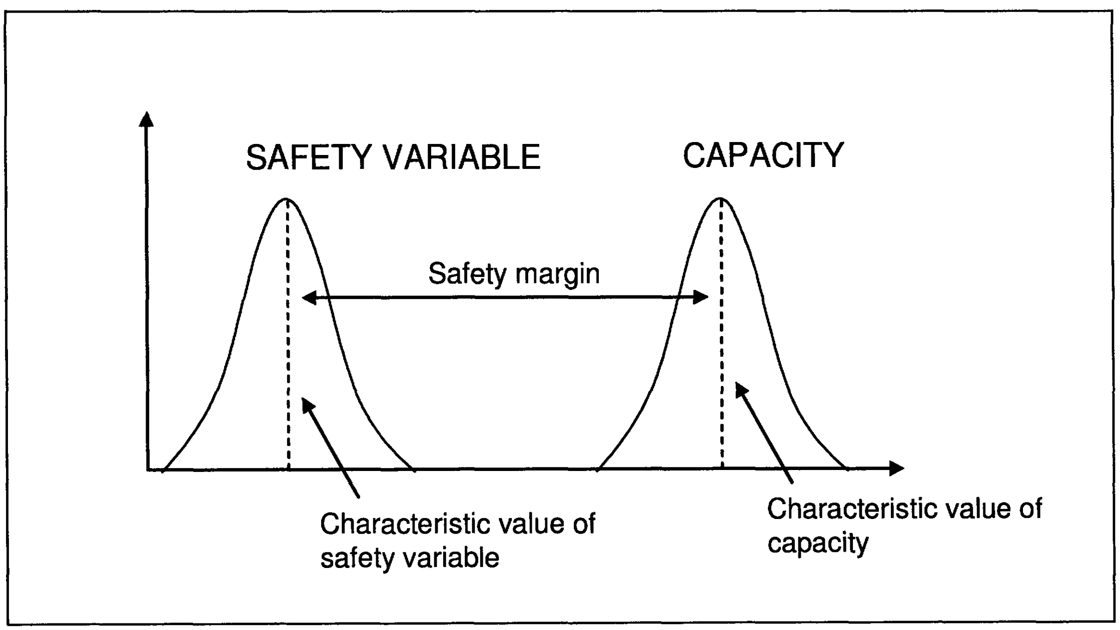

Figure 2: Definition of safety margin

The capacity is often uncertain, and the regulator may choose to define a regulatory limit well below the capacity. The definition of safety margin can then be defined as the sum of the design margin and the regulatory margin. The design margin is the difference between the regulatory limit and the characteristic value of the safety variable; and the regulatory margin is the difference between the characteristic value of the capacity and the regulatory limit.

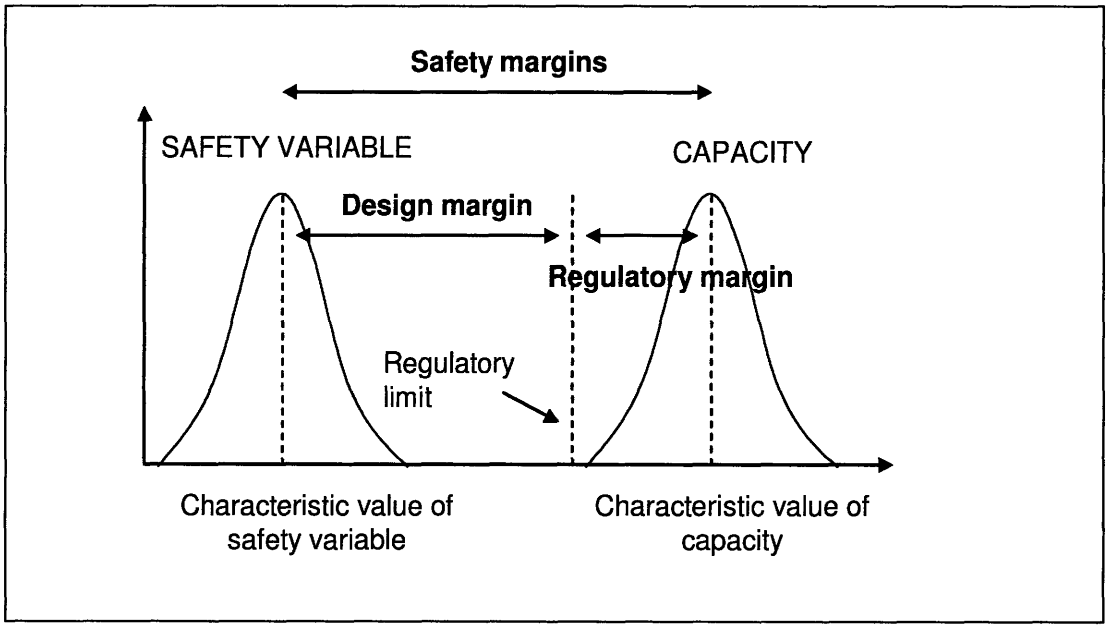

Figure 3: Design and regulatory margins (USNRC, 2006)

To calculate the characteristic value of the load: one can either do a best estimate calculation using realistic codes and analyses, or one can make conservative assumptions to calculate a value, that if below the acceptance criterion, ensure that adequate safety margin is provided without having to quantify it. Table 1 summarizes the characteristics of the conservative and best estimate approaches.

Applied codes	Input & Boundary and initial conditions	Assumptions on system availability	Approach
Conservative codes	Conservative input	Conservative	Deterministic
Best estimate (realistic) codes	Conservative input	Conservative	Deterministic

Best estimate codes + Uncertainties	Realistic input + uncertainties	Conservative	Deterministic
Best estimate codes + Uncertainties	Realistic input + uncertainties	PRA-based	Deterministic + Probabilistic

Table 1: "Conservative approach" versus "Best estimate approach" (IAEA, 2001)

Adequate safety margins are currently ensured by requiring that the conservative value for the safety variable be below the regulatory value for the capacity. For instance, 10 CFR Part 50.46 stipulates that the peak clad temperature during transients for a Light Water Reactor (LWR) cannot exceed 2200°C during a LOCA. The designer uses conservative assumptions to ensure that this requirement is met. The "real" safety margin is not quantified. Research efforts are currently undertaken to improve computer codes to allow best estimate calculations and uncertainty analyses.

III.C. The trend towards risk-informing regulations

III.C.1. Defining risk quantitatively

Risk analysis is the discipline that has the objective of capturing risk by answering three questions (Kaplan and Garrick, 1981): (1) what can happen?, (2) how likely is it that it will happen?, and (3) if it does happen, what are the consequences? A risk analysis consists therefore in identifying all the possible scenarios, for which both the probability and consequences are assessed.

The *"level-1"* definition of risk by Kaplan and Garrick is a set of triplets that express for each possible outcome its probability and consequence: R=$\{\langle S_i, P_i, X_i\rangle, i = 1,2...N\}$, with P_i being the probability of the scenario S_i, and X_i the measure of damage or consequence measure of the scenario. The integration of uncertainties leads to the *"level-2"* definition of risk.

There are several ways to display the risk of a system. Among them are risk-curves, which express the frequency of exceeding a certain consequence (Complementary Cumulative Density

Function). Epistemic uncertainties can be displayed on risk-curves (Figure 4). These curves have to be read vertically: the frequency of exceeding a certain consequence is uncertain and the different confidence levels for the frequency can be read vertically, as shown on the following figure.

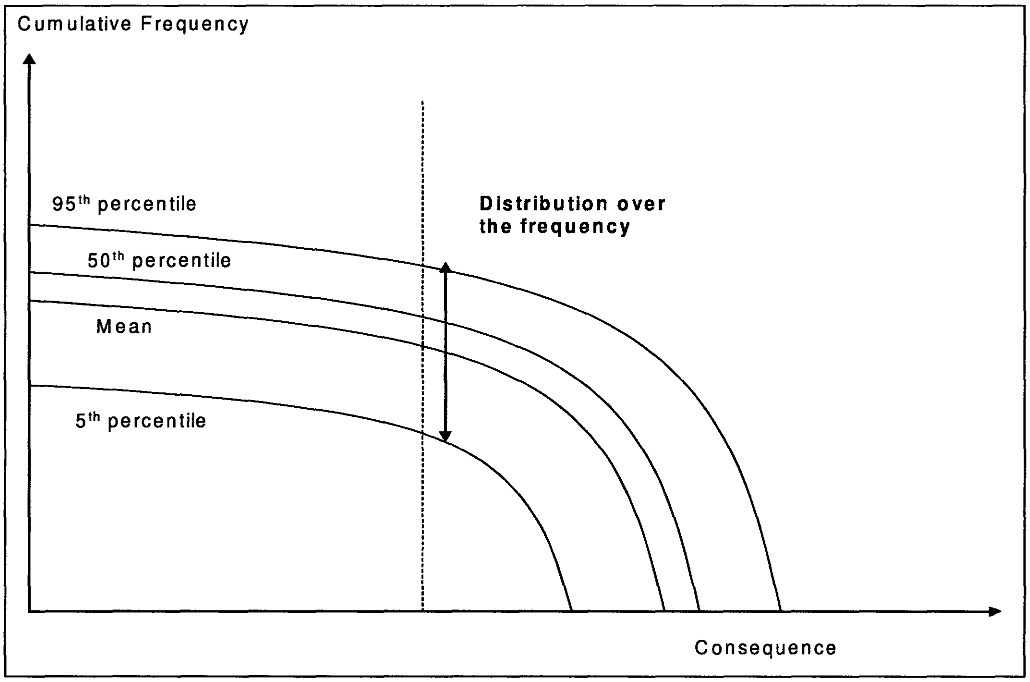

Figure 4: Display of uncertainties on F-C curves

Probabilistic Risk Assessment (PRA) is an analytical technique for systematically identifying potential outcomes of a known initiating event. Major PRA studies include the 1975 Reactor Safety Study and the 1990 NUREG-1150 study, which assessed the risk of severe accidents for five nuclear power plants. There are several levels of PRA:

- Level-1 PRAs quantify the frequency of having core damage (CDF);
- Level-2 PRAs quantify the frequency of a large early release of radioactive material (LERF). Figure 5 illustrates the different items that need to be assessed and quantified for a level-2 PRA;
- Level-3 PRAs calculate the off-site consequences of potential accidents. This latter level is the most uncertain since it requires the modeling of radioactive plume dispersion and the modeling of health effects.

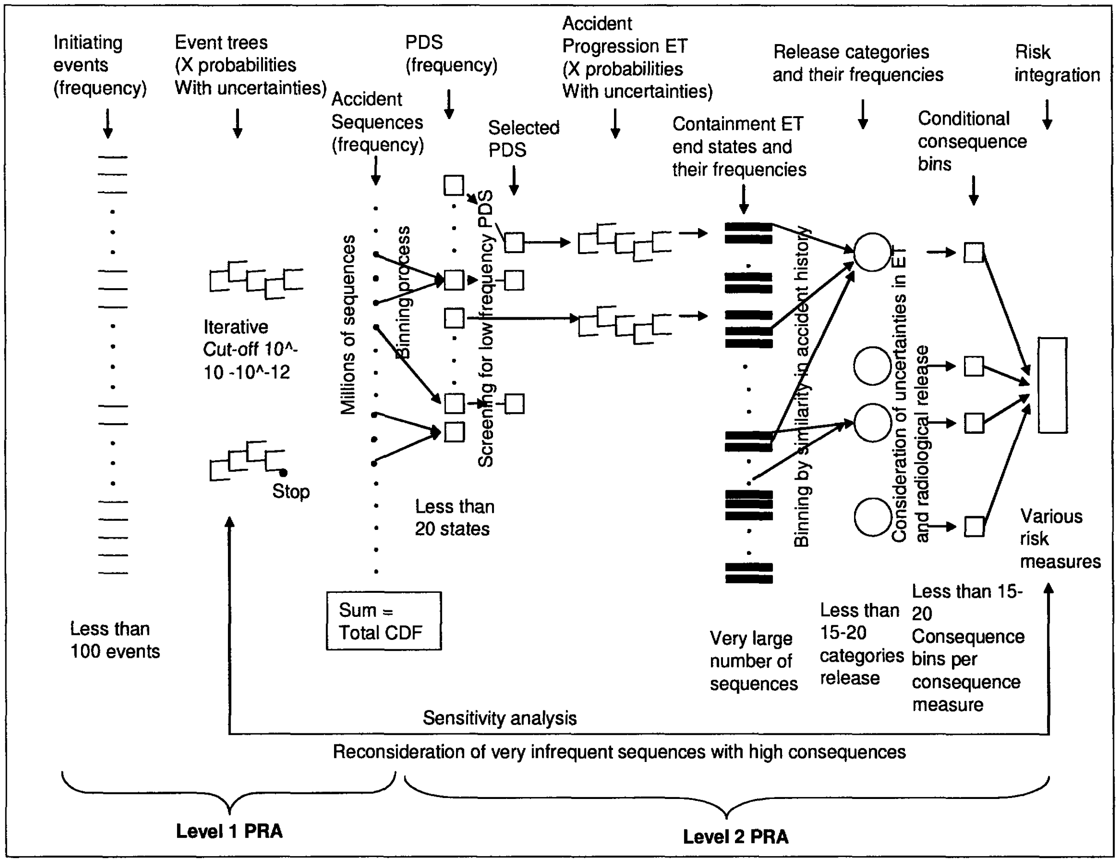

Figure 5: PRA modeling

(Cazzoli et al, 1993)

III.C.2. Quantification of risk in the regulations

In parallel to the maturation of risk assessment tools, the USNRC started quantifying risk acceptance criteria. In 1986, the USNRC issued the Safety Goal Policy Statement (USNRC, 1986), in which it stated what it judged to be an acceptable level of risk from nuclear power plants. Two Quantitative Health Objectives (QHOs) were defined:

> "The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
> The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of

one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes."

In August 1995, the Commission issued a policy statement on the use of PRA methods in nuclear regulatory activities (USNRC, 1995). The policy statement recommended that the use of PRA technology be "increased in all regulatory matters in a manner that complements the USNRC's traditional defense-in-depth philosophy." It also recommended that PRA and associated analyses be used to reduce unnecessary conservatism associated with current regulatory requirements and guides, license commitments, and staff practices, in order to focus the regulatory actions on where the risk is the highest.

Significant change has been introduced in the past decades in the regulations and we can observe an increasing reliance on risk quantification. However, there are still wide parts of the regulations, such as licensing requirements, that haven't benefited fully from PRA insights.

III.D. Justifications for a new licensing approach

There are three types of issues associated with the current licensing approach:

- *Current regulations focus on LWR design.* This issue has already been raised at the USNRC for reactor technologies such as the Pebble Bed Modular Reactor (PBMR), for which risk metrics such as the Core Damage Frequency might not be applicable. Getting exempted from LWR requirements is a long process, which necessarily creates unpredictability in licensing and might discourage investment in new reactor designs.

- *Deterministic requirements may cause unnecessary burden and may miss critical safety issues.* Regulators placed additional barriers and imposed new requirements asking the question: What if we are wrong? What if barriers fail? This led to the addition of safety features that did not necessarily increase plant safety. For instance, the "Reactor Safety Study: An assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", known as WASH-1400 (USNRC, 1975), found that small LOCAs and transients were dominant contributors to the risk of a plant, contradicting the previous purely deterministic approach that only considered very large pipe breaks in the reactor coolant system.

- *Unnecessary requirements may be very costly and therefore are a major drawback to nuclear power development.* In an article entitled "Who Killed U.S Nuclear Power?" Marsha Freeman, associate editor of the magazine 21st Century Science Technology, points out the role of nuclear regulatory actions in the seventies: "Billions of dollars were spent by nuclear utilities to retrofit plants for increased safety, much of which retrofitting was known by many in the industry to be unnecessary" (Freeman, 2001). Charles Komanoff, an energy economist and environmental activist, released a study in 1981 (Komanoff, 1981) proposing that the real cost in constant "steam-plant" dollars per kilowatt to complete nuclear power plants in the United States increased by 142% from the end of 1971 to the end of 1978, taking into account the inflation in the costs of standard construction inputs such as labor, equipment, and materials.

  Note however that quantification of the role of regulations on cost increases is a difficult task and studies are scarce. Nuclear power plants are very complex systems, which makes it difficult to directly relate one regulation to an increase in costs. However, even if figures are exaggerated, most experts agree that the tremendous increase in requirements has had a very strong impact on costs, while not all the new requirements were justified.

  George Apostolakis, chairman of the ACRS PRA subcommittee, states that PRA has a great role to play regarding "the regulatory burden that was created in some instances, such as in quality assurance requirements" (Apostolakis, 2000). He further says that "one utility has indicated that if it implemented graded quality assurance guidance, its savings would be up to $ 2 million a year". Regulatory Guide 1.176 "An Approach for Plant-Specific, Risk-Informed Decision making: Graded Quality Assurance" provides guidance on how to risk-inform the regulations and requires quality assurance adjusted to the level of safety needed.

III.E. Conclusion

The objective of the framework drafted by the USNRC is to produce a risk-informed and technology-neutral licensing process, which constitutes a major change in the regulations since it calls for an increase reliance on risk assessment tools while maintaining a high-level Defense-In-Depth. Such task represents a tremendous challenge for both the regulator in charge of defining the process and the industry, which will have to comply eventually.

F-C curves are a good example of the combination of probabilistic and deterministic principles: They allow both a quantification of risk and the implementation of structuralist and rationalist Defense-In-Depth through the quantification and implementation of safety margins.

Part IV. Use of frequency-consequence curves in new reactor licensing

Regulations of nuclear power plants have focused until now on Light Water Reactors and have not systematically incorporated insights and benefits from probabilistic risk assessment methods. With the goal of risk-informing the regulations and making the licensing process more efficient, predictable and stable for advanced reactors, the US Nuclear Regulatory Commission has recently drafted a technology-neutral framework for new plant licensing. The new licensing rules would be applicable to Generation IV commercial nuclear power plants only, and would constitute an alternative to 10 CFR Part 50. The current working draft released by the USNRC in August 2006 (USNRC, 2006) envisions two major uses of F-C curves: a tool to ensure implementation of the USNRC's safety expectations as well as a tool to identify and select the Licensing Basis Events (LBEs), intended to replace the Design-Basis Accidents (DBAs).

The objective of this part is two-fold: first, present the F-C curve concept proposed by the USNRC, and, second, understand the extent to which Licensing Basis Events constitute an improvement over Design-Basis Accidents.

IV.A. Expected level of safety for future plants

The level of safety that new plants are expected to meet, captured by the framework, has been defined in the policy statement on the regulation of advanced nuclear power plants (USNRC, 1994), in which the Commission has expressed two expectations:

- That advanced nuclear power plants will show enhanced margins of safety.
- That advanced reactor designs will comply with the Commission Safety Goal Policy Statement, i.e. that plants will comply with the Quantitative Health Objectives.

A three-region approach to risk acceptability has been developed. The requirements developed through the framework will ensure that the risk lies in the lower region, and that there is only a small chance that the risk can be in the intermediate region, and a negligible probability that it lies in the unacceptable region.

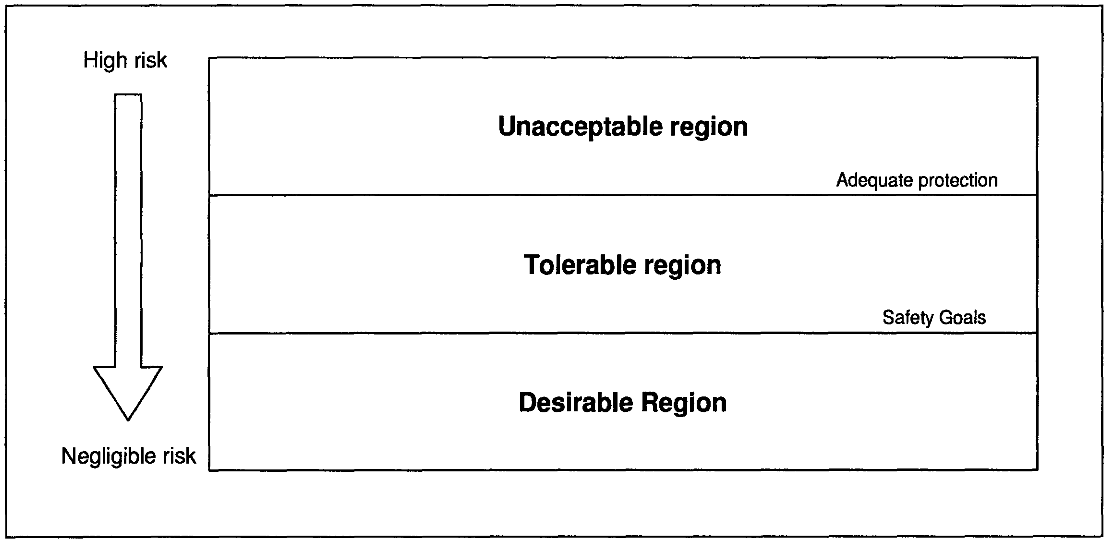

Figure 6: Three Region Approach to Risk Tolerability

IV.B. Definition of frequency-consequence curves in the framework

The F-C curve proposed for use by the Commission's staff relates the frequency of potential accidents to acceptable radiation dose released by these potential accidents for an individual at the site boundary. The underlying principle is that the higher the consequence of an event, the lower the frequency of the event must be. The F-C curve is derived from current regulatory requirements that can be found in 10 CFR Parts 20, 50 and 100.

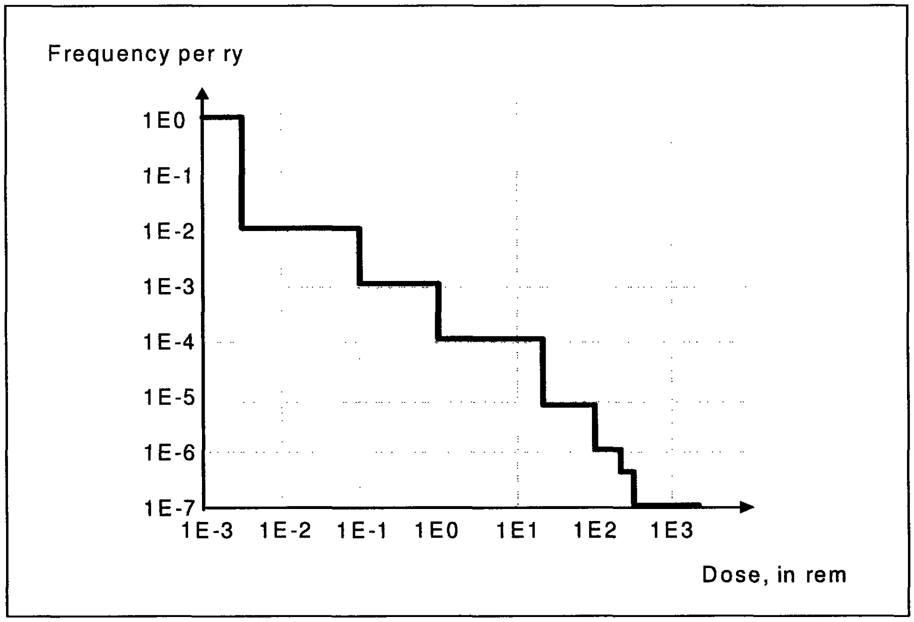

Figure 7: Frequency-consequence curve proposed in the USNRC framework

As an example, 10 CFR § 20.1301 specifies the dose limits for individual members of the public:

> "Each licensee shall conduct operations so that the total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year".

Therefore, events resulting in doses of 100 mrem shouldn't have a frequency above 1. This is translated on Figure 7 by limiting the frequency of events resulting in doses of 5-100 rems to 0.01/ry. The figure presents the F-C curve proposed for use by the USNRC as of August 2006. One should note that 10 CFR § 20.1301 specifies a limit on the integrated risk, not from a single event; whereas the interpretation done for the F-C curve is on a single event basis.

IV.C. Use of the frequency-consequence curve to implement USNRC's high-level safety expectations

IV.C.1. Each event sequence must lie individually below the F-C curve

A PRA has to be completed (whose technical requirements are detailed in the framework). The PRA encompasses all internal and external events as well as all modes of plant operation. The PRA is used to generate a sufficiently complete set of accidents scenarios, whose frequencies and consequences are calculated with uncertainties accounted for: all accident sequences are identified in terms of a distribution of their frequencies and end states.

To implement USNRC's high-level safety expectations, each event sequence, defined by its mean frequency and mean consequence dose, must individually lie in the lower region of the F-C curve.

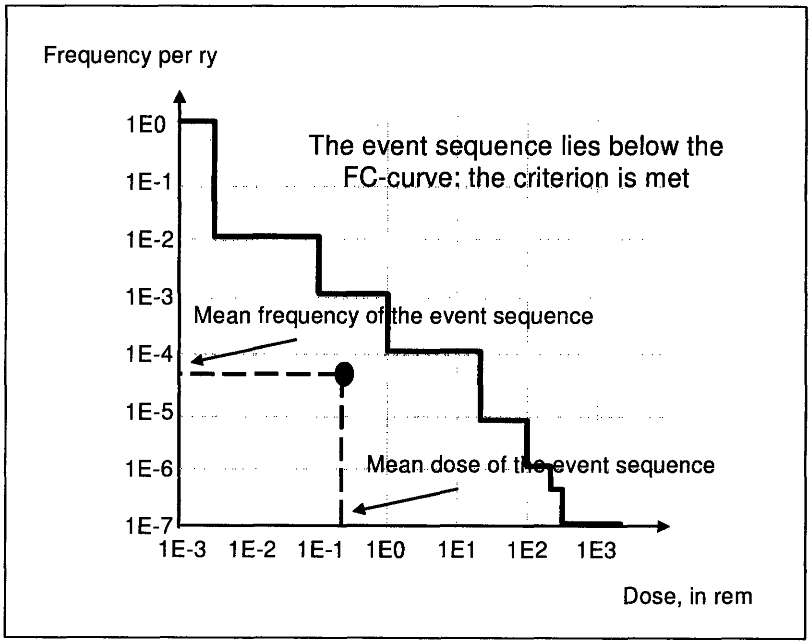

Figure 8: Criterion for individual event sequence

IV.C.2. The integrated risk is not assessed with the frequency-consequence curve

Each PRA sequence must meet individually the criterion imposed by the F-C curve on a mean value basis, which implies that each sequence meets individually the QHOs. However, the overall risk is not captured by the F-C curve and the PRA results must also demonstrate that the total integrated risk over all accident sequences satisfy both QHOs.

To show compliance with the QHOs, a level-3 PRA is needed unless surrogates objectives can be determined. For now, no surrogates similar to CDF and LERF have been defined on a technology-neutral basis for advanced reactors. However, even if surrogates were to be defined, sufficient experience and time would be needed in order to have confidence in their use.

IV.D. Use of frequency-consequence curves to identify and select Licensing Basis Events

IV.D.1. Design-Basis Accidents

The current fleet of U.S. reactors was licensed using a deterministic approach, which evolved from the licensing of the first commercial power plant on a case-by-case basis to the emergence in the mid sixties of generic criteria that the reactor design must meet.

A deterministic approach refers to the principle of "determinism", which holds that:

> "Specific causes completely and certainly determine effects of all sorts. As applied in nuclear technology, it generally deals with evaluating the safety of a nuclear power plant in terms of the consequences of a predetermined bounding subset of accident sequences" (USNRC Glossary, 2006).

Hence, the deterministic approach relies on the concept of design-basis accidents, which are postulated accidents

> "that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to assure public health and safety" (USNRC Glossary, 2006).

These accidents envelop the whole spectrum of accidents. If a power plant is able to withstand the design-basis accidents, which assume worst-case scenarios, then it is able to cope with all accident challenges:

> "The design basis accidents were not intended to be actual event sequences, but rather, were intended to be surrogates to enable deterministic evaluation of the response of a facility's engineered safety features. These accident analyses are intentionally conservative in order to compensate for known uncertainties in accident progression, fission product transport, and atmospheric dispersion" (USNRC Regulatory Guide 1.183, 2000).

Beyond design-basis accidents are, on the other hand, accident sequences that are possible, but are not fully considered in the design process because they are judged to be too unlikely. The redundancy of systems and extensive implementation of margins arising from Design-Basis Accident evaluations have resulted in plant designs that have considerable robustness and capability to mitigate potential severe accident scenarios (USNRC, 2004).

IV.D.1.a. The General Design Criteria

All the Light Water Reactors (LWRs) conceived and proposed to the AEC for construction permits, from the Shippingport reactor in 1953 to Dresden 2 in 1965, were generated without a set of safety criteria that the design must meet.

All Light Water Reactors (LWRs) conceived and proposed to the AEC for construction permits from the Shippingport reactor in 1953 to Dresden 2 in 1965 were generated without a set of safety criteria that the design must meet.

Prior to 1965, the individual design criteria evolved over the years on a case-by-case basis. New criteria were introduced as the result of rector-specific or site-specific issues and tended to emerge from questions about low-probability events not previously considered, or from unusual operating experience with generic implications.

In 1965, the AEC staff started developing general design criteria. The original criteria were revised in 1967 and again in 1971 when the AEC published a general set of design criteria that became Appendix A to 10 CFR Part 50.

Appendix A to 10 CFR 50 states that:

> "An application for a construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.
> These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission"

One of the most famous criteria is the single failure criterion. Appendix A to 10 CFR Part 50 defined "single failure" as:

> "An occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions."

Other criteria, such as criterion 35 on the Emergency Core Cooling System (ECCS) refer to the concept of single failure. Criterion 35 states that the emergency cooling system should be designed to withstand a postulated Loss of Coolant Accident (LOCA) defined as double-ended rupture of the largest pipe of the reactor coolant system, the concurrent loss of offsite power, and a single failure of an active EECS component in the worst possible place.

IV.D.1.b. Design-basis accidents as a tool to show compliance with licensing requirements

The design-basis accidents stem from the General Design Criteria.

10 CFR 50.34 requires that each application for a construction permit for a nuclear reactor facility include a Preliminary Safety Analysis Report (PSAR) and that each application for a license to operate such a facility include a Final Safety Analysis Report (FSAR). Section 50.34 specifies in general terms the information to be supplied in these Safety Analysis Reports (SARs). Regulatory Guide 1.70 describes in more details the information that should be provided in the SAR. Chapter 15 of the Safety Analysis Report focuses on accident analyses and provides guidance on the classification of events and on the methodology that should be used. As mentioned earlier, the applicant must show that its design conforms to the General Design Criteria and that the plant is able to withstand the postulated design-basis accidents

For instance, an applicant can postulate a LOCA inside the containment, assuming a worst case of piping break in order to represent an envelope evaluation for liquid or steam line failure inside the containment. The assumptions and calculations should be conservative.

As an example, Regulatory Guide 1.3 indicates what type of conservative assumptions should be done (e.g. "infinite cloud" assumption). Regulatory Guide 1.70, that provides guidance for the SAR, acknowledges that "there may be instances in which the applicant will not agree with the conservative margins inherent the design basis approach approved by the USNRC staff" and in which the applicant might want to do a realistic analysis. The applicant may present his analysis but he is reminded that "the known USNRC assumptions should nevertheless be used in the design basis analysis."

IV.D.1.c. The "Maximum Credible Accident" concept

Another postulated accident, which plays a fundamental role in the licensing process, is the "Maximum Credible Accident", postulated for siting purposes.

According to David Okrent, former ACRS member and author of a book on the history of the regulatory process (Okrent, 1981), the principle was mentioned by Clifford Beck, member of the regulatory staff, in a nuclear congress in Rome in 1959. The philosophy behind Design-Basis Accidents was summarized then as follows:

> "If the worst conceivable accidents are considered, no site except one removed from population areas by hundreds of miles would offer sufficient protection. On the other hand, if safeguards are included in the facility design against all possible accidents having unacceptable consequences, then it could be argued that any site, however crowded, would be satisfactory... assuming of course that the safeguards would not fail and some dangerous potential accidents had not been overlooked. In practice, a compromise position between these two extremes is taken. Sufficient reliance is placed on the protective features to remove most of the concern about the worst conceivable accidents, though there is seldomly sufficient confidence in the facility safeguards to be sure that all hazards have been eliminated. Thus, a possible reactor site is reviewed against the possibility of credible accidents, and their consequences, which might occur despite the safeguards present.
> It is inherently impossible to give an objective definition or specification for "credible accidents" and thus the attempt to identify these for a given reactor entails some sense of futility and frustration, and further, it is never entirely assured that all potential accidents have been examined."

Clifford Beck, in this speech, puts the emphasis on the difficulty of defining credible accidents and on the need for additional barriers due to lack of knowledge (epistemic) uncertainties, laying the ground for the concept of defense-in-depth.

Following up on this idea in 1961, the AEC, under the leadership of Clifford Beck, published for comment in the Federal Register, siting criteria that included concepts such as a low-population zone, an exclusion area, and a population center distance:

> "For purposes of site evaluation, an accident was postulated in which the noble gases and half the radioiodine were released to a containment building that was assumed to maintain its integrity, and in which guideline doses of 25 rem whole body and 300 rem to the thyroid were not to be exceeded under the specified conditions. This postulated accident (the maximum credible accident or MCA) whose consequences were not to be exceeded by any credible accident, became the focus of siting evaluation. [...] Most safety improvements which developed were related to meeting the requirements of the postulated MCA."(Okrent, 1981)

The use of postulated accidents to show compliance with siting requirements is still in the regulations. Section 100.11 of 10 CFR Part 100 provides criteria for determining the Exclusion Area, Low Population Zone, as well as the Population Center Distance. To evaluate a proposed site, the applicant should assume a fission product release, "based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."

IV.D.1.d. The difficulty of dealing with incredible accidents

Although the notion of "credibility" seems to refer to the concept of likelihood and probability, expert judgment and experience were the basis for defining credible accidents. The question of how to deal with incredible accidents has always been a thorny issue. It is important to note that the Maximum Credible Accident was assumed to be contained.

The difficulty of dealing with incredible accidents can be illustrated by the question of the reactor pressure vessel integrity that arose in 1965 (Okrent, 1981). The AEC regulatory staff was unwilling to consider accidents it qualified as incredible. The issue of the integrity of the reactor pressure vessel had been raised several times by the ACRS before 1965. For instance, in a 1961 report to the AEC, it recommended the development of adequate codes and standards for the pressure vessel and other parts of the primary systems of power plants. However, failure of the reactor pressure vessel was considered as "incredible" for the LWR and BWR reviewed before 1965. No protection against gross vessel failure was provided, even though the possible consequences of such failure would have potentially led to a major uncontrolled release of radioactivity.

The issue was especially complex since there did not seem to be clearly feasible way to prevent core melt and ensure containment integrity in case of a catastrophic pressure vessel failure. At an ACRS subcommittee meeting dedicated to the Dresden II reactor licensing application, the vendor representative, asked about the consequences of a potential pressure vessel, replied, "The containment could withstand a larger break than the maximum credible accident but not a complete break of the pressure vessel." In November 1965, the ACRS recommended in a letter to the AEC that some provisions be made against the unlikely accidents and that means be developed to ameliorate the consequences of a major vessel pressure rupture (Okrent, 1981).

In a 1967 paper presented to the IAEA, Farmer criticized the approach taken in differentiating credible accidents from incredible ones:

> "No engineering plant and no structure is entirely risk free, and there is no logical way of differentiating between credible and incredible accidents. The incredible is often made up of a combination of very ordinary events – for example the breakdown or the deterioration that occurs in normal plants and their measuring instruments – and the credible may actually be exceedingly improbable. The logical way of dealing with this situation is to seek to assess the whole spectrum or risks in a quantity-related manner" (Farmer, 1967).

IV.D.1.e. Licensing Basis Events should replace Design-Basis Accidents

Design Basis-Accidents (DBAs) are inherited from a purely deterministic approach to safety. Furthermore, they might not be applicable anymore to reactors different from LWRs. They must be therefore replaced.

The identification and selection of Licensing Basis Events (LBEs) is a fundamental difference between the previous licensing process and the one proposed in the new framework. LBEs are accidents that must be considered in the plant safety analysis and that represent a challenge to safety. They play a role in the licensing process similar to the DBAs, for they provide assurance that the design meets various accident challenges with adequate margins. However, LBEs encompass a much broader range of events since they also include, for instance, some events that do not involve radioactive release. There are two ways of selecting LBEs: a *probabilistic* selection from the PRA sequences, as well as a *deterministic* selection process that ensure that all uncertainties are accounted for. LBEs are chosen so that their aggregate represents the whole frequency range of the F-C curve.

IV.D.2. Probabilistic selection of Licensing Basis Events

The probabilistic selection process of LBEs uses the results of the full scope PRA: once all the PRA sequences have been defined in terms of a distribution of their frequencies and end states, LBEs can be selected from the PRA sequences

IV.D.2.a. Steps for Licensing Basis Event selection

The PRA is first modified so as to credit the mitigating functions that are to be considered safety-significant: indeed, any function and the associated Systems, Structures, and Components (SSCs) included in the PRA and used to define the LBEs is considered safety-significant unless guaranteed failure has been assumed. The selection process of LBEs is as follows:

- The point estimate frequency for each resulting event sequence of the modified PRA is calculated. Only the event sequences with a point estimate frequency above 10^{-8} /ry are eligible for the LBE selection process.

- The mean and 95th percentile for all event sequences remaining is determined, and all the event sequences whose 95th percentile frequency is below 10^{-7} are screened out.
- Similar accident sequences, defined as sequences that "have a similar initiator and display similar accident behavior in terms of system failures and/or phenomena and lead to similar source terms" are then grouped together in event classes.
- For each event class, the event sequence with the bounding consequence is selected. The selected event sequence defines the accident behavior and consequence.
- Then, for each event class, the LBE frequency is determined by setting the LBE's mean frequency equal to the highest mean frequency of the event sequences, and the 95th percentile equal to the highest 95th percentile frequency. The parameters of LBEs are illustrated on Figure 9:

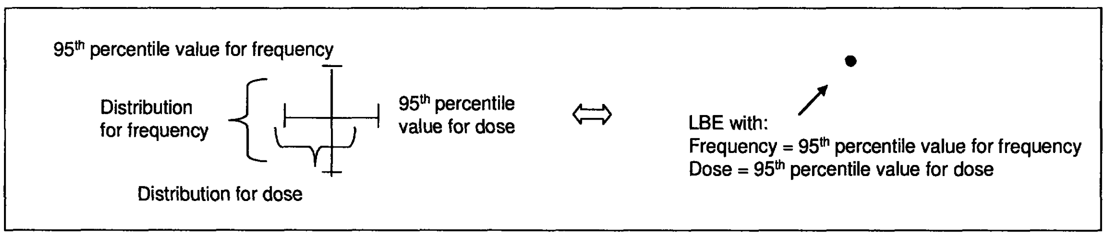

Figure 9: Parameters of Licensing Basis Events

One should not that such process might be difficult to implement for the highest event frequency category, since there is no release of radioactivity. Therefore, engineering judgment may be used.

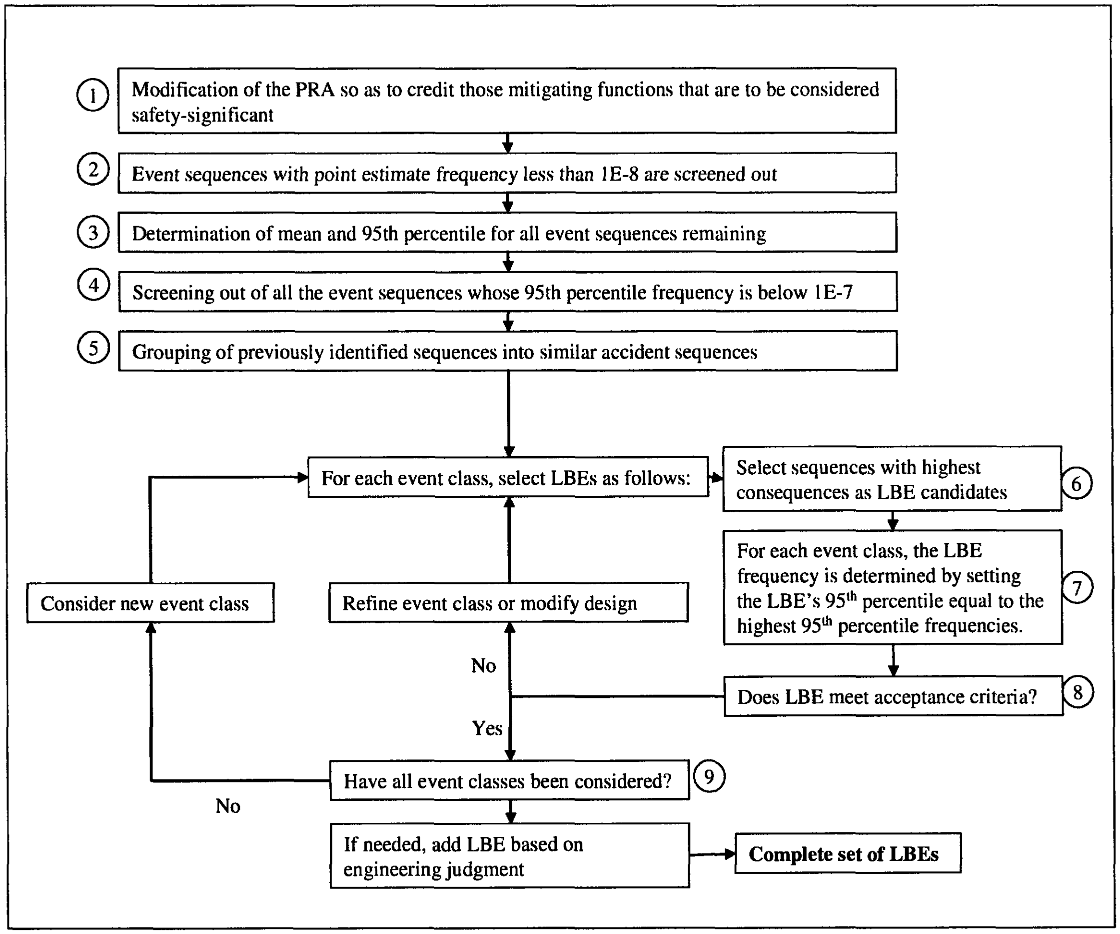

Figure 10: Selection process of Licensing Basis Events

IV.D.2.b. Criteria for selected Licensing Basis Events

Each selected LBE has to lie below the F-C curve.

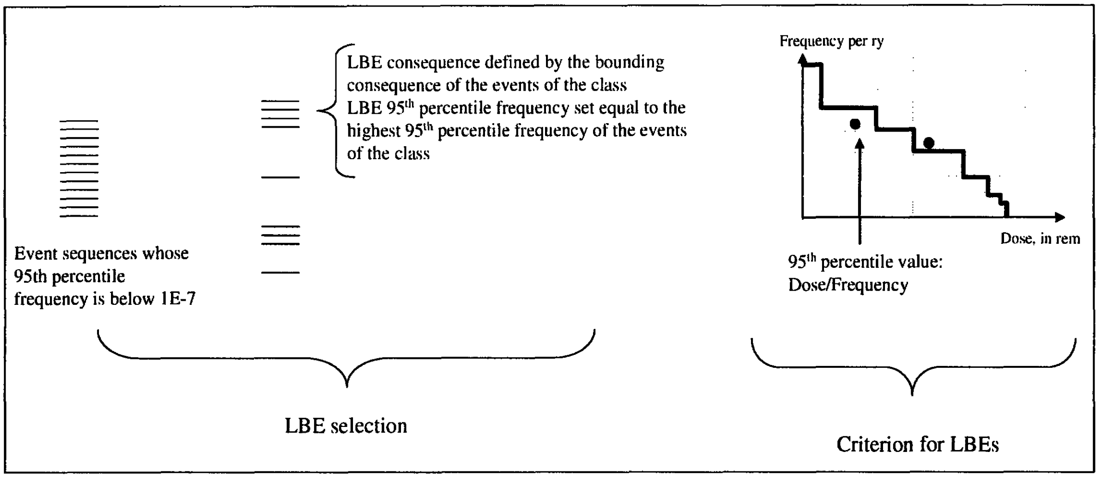

Figure 11: Criterion for probabilistically-selected LBE

Furthermore, for defense-in-depth purposes, LBEs must meet additional deterministic criteria. For that purpose, the region below the F-C curve is divided into three frequency regions, as shown in Figure 12. The rationale for such division is summarized in Table 2. The principle is that it is desirable to have more stringent deterministic criteria for frequent events, than for less frequent events.

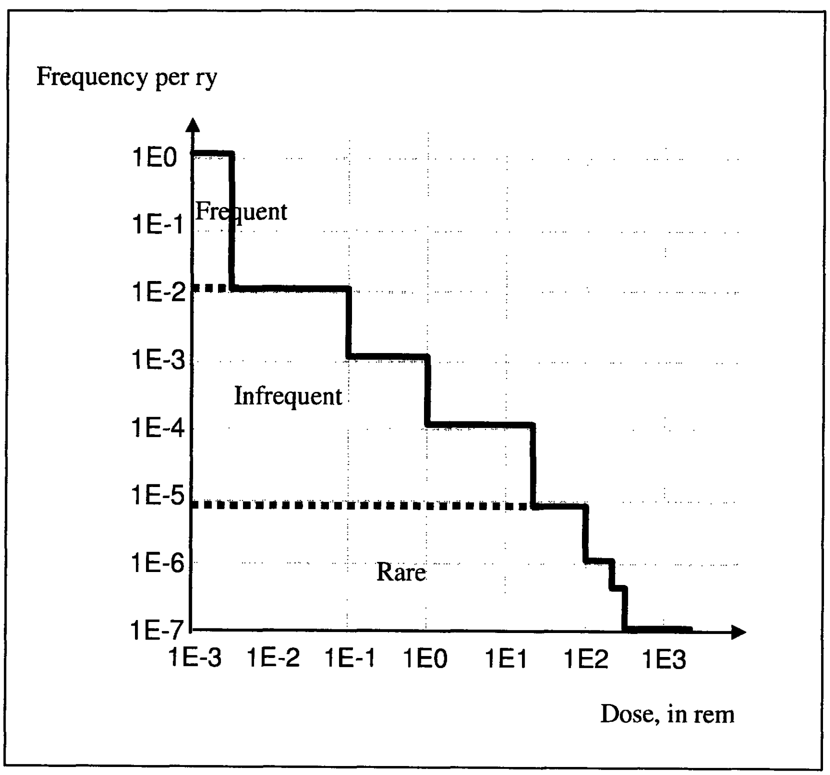

Figure 12: 3-region approach

Category	Frequency	Basis for choice
Frequent	$> 10^{-2}$ /ry (mean value)	Captures all event sequences expected to occur at least once in lifetime of a plant, assumed to be 60 years
Infrequent	$< 10^{-2}$ /ry and $> 10^{-5}$ /ry (mean value)	Captures all event sequences expected to occur at least once in lifetime of population of plants, assumed to be 1000
Rare	$< 10^{-5}$ /ry and $> 10^{-7}$ /ry (mean value)	Captures all event sequences not expected to occur in lifetime of the plant population, but needed to assess the Commission's safety goals

Table 2: Classification of event sequences according to their mean frequency

The previous table applies to event sequences, not only initiating event (IE) frequencies. The framework suggests that each applicant propose cumulative limit on IE frequencies for each of the LBE frequency event categories (for instance, the initiating events with potential to defeat two or more protective strategies should have a frequency below 10^{-7} per plant year). The USNRC and the applicant must agree upon the cumulative IE frequency, taking into account the design characteristics. The limits are monitored on the long term by a living PRA.

The LBEs, based on their frequency category must meet additional deterministic criteria.

Category	Frequency	Deterministic criteria
Frequent	$> 10^{-2}$ /ry (mean value)	- No impact on the safety analysis assumption occurs - No barrier failure occurs - Redundant means of reactor shutdown remain functional - Redundant means of decay heat removal remain functional
Infrequent	$< 10^{-2}$ /ry and	- A coolable geometry is maintained

	$> 10^{-5}$ /ry (mean value)	- At least one barrier remains - At least one means of reactor shutdown remains functional - At least one means of decay heat removal remains functional
Rare	$< 10^{-5}$ /ry and $> 10^{-7}$ /ry (mean value)	No additional deterministic criteria

Table 3: Additional deterministic criteria depending on frequency category

Furthermore, depending on the frequency category, LBEs must satisfy additional dose criteria. For instance, in the higher event frequency category, the cumulative dose has to be below the 5mrem dose specification of 10 CFR 10 Appendix I.

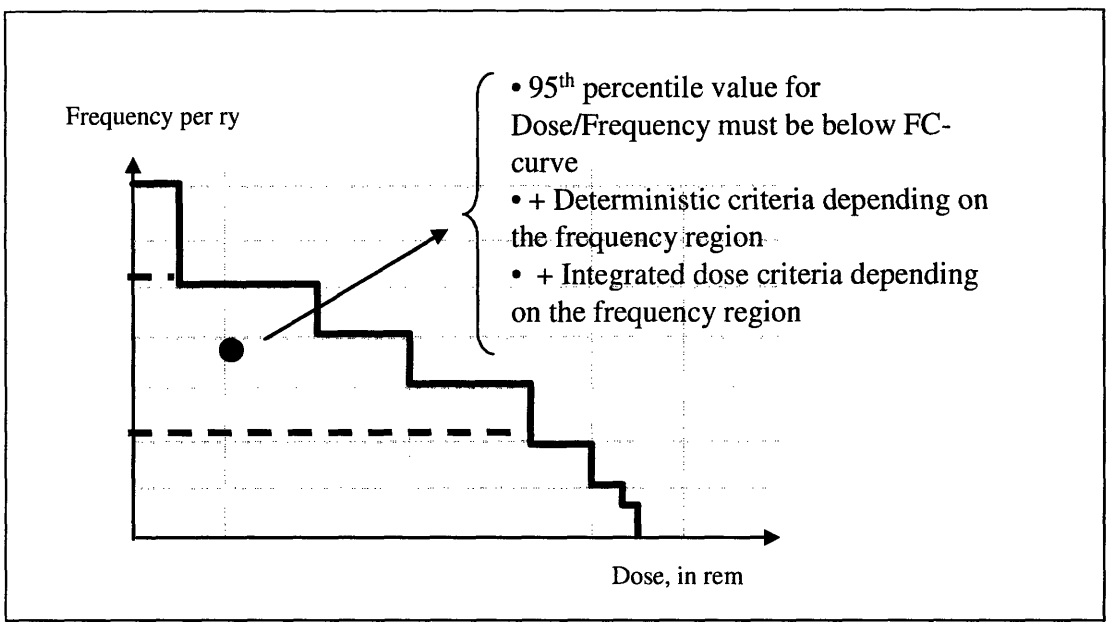

Figure 13: Summary criteria for LBEs

IV.D.3. Deterministic selection of Licensing Basis Events

For siting concerns, one LBE has to be postulated deterministically so as to prove that, regardless of the features incorporated in the plant to prevent an unacceptable release of radioactive material from the fuel and the reactor coolant system, there are additional ways to prevent an unacceptable release to the public. This LBE has to be analyzed mechanistically using conservative assumptions. This event is the event postulated in 10 CFR 100.

IV.E. Analogy between the current framework draft and Farmer paper

In 1967, Farmer proposed to use a F-C curve in order to assess the risk of a power plant from a siting perspective. The F-C curve proposed by the USNRC presents some analogy with the Farmer curve. In both approaches, accident sequences are first analyzed using risk quantification techniques and their acceptability is assessed on a frequency-consequence diagram, based on the similar principle that the higher the consequence of an event sequence is, the lower its frequency must be:

> "A measure of risk can be obtained by estimating the probability of the failure and assessing the consequences. Any initiating event – for example, failure of piping, delays in the operation of control systems, loss of circulator power, or combinations of these – can set up an accident sequence that can follow many paths [...] The full safety evaluation then comprises a spectrum of events with associated probabilities and associated consequences". (Farmer, 1967)

However, if both curves present many similarities, the consequences considered are highly different. Indeed, the Farmer paper addresses siting problems in the sense it limits for each event sequence the total amount of radioactive ^{131}I released. Therefore, Farmer addresses societal risk. The USNRC draft addresses individual risk, i.e. the dose for an individual at the site boundary.

IV.F. Improvements due to Licensing Basis Events

The definition and use of LBEs contributes greatly to the definition of a technology-neutral and risk-informed licensing process. Several improvements should be noted:

- *Calculations to obtain the distribution of frequency and dose are realistic*; except for the source term calculated using the 95th value of the probability range for the amount of radionuclides released. Distributions on the frequency and the dose are assessed.

- *The probabilistically selected LBEs contribute to the existence of quantifiable safety margins.* Regulatory limits on the frequency and consequence of potential events are set by the F-C curve so that adequate regulatory margin is provided. There is indeed a lot of uncertainty regarding the health effects caused by defined radioactive doses, which calls for a conservative regulatory limit. The designer can define additional design margins (distance between a calculated value for the safety variable and the regulatory value, Figure 14).

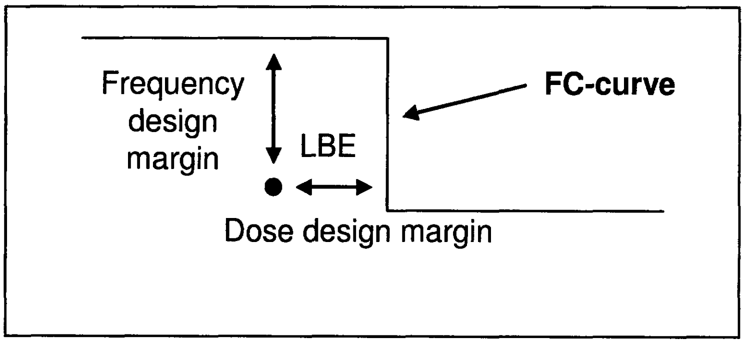

Figure 14: Licensing Basis Event Margins

- *The framework allows a performance-based approach*: indeed, the designer may choose to add deterministic LBEs based on his judgment, but this is not required beforehand by the regulator.

- *Defense-In- Depth (both structuralist and rationalist) remains a fundamental principle of the approach:* LBEs must satisfy certain fundamental criteria depending on their frequency (rationalist and structuralist Defense-In-Depth), and the postulated accident for siting purposes ensures that a balance between prevention and mitigation is maintained (structuralist Defense-In-Depth). Implementation of safety margins ensures that uncertainties are adequately coped with.

IV.G. Conclusion

F-C curves are powerful risk assessment tools, for they provide enhanced safety margins and a rational way to define Licensing Basis Events.

USNRC's use of F-C curves is quite innovative, since these tools are classically used to assess societal risk as opposed to individual risk. The question of including societal risk in the regulations has been regularly raised and it is legitimate in the context of the new framework to ask if societal risk should be included in the new licensing approach, and how could F-C contribute to societal risk assessment.

Part V. What is societal risk?

Before considering the use of F-C curves for societal risk assessment, we will define in this part what societal risk is, and introduce the general concepts attached to it, for instance multiple-fatality aversion. In the more specific context of nuclear power plants, we will identify three main sources of societal risk increase: a degradation of the plant safety, an increase of the core inventory, which can in turn increase the risk to the population as a whole, and an increase in the number of people living around the plant.

V.A. What is societal risk as opposed to individual risk?

V.A.1. Individual risk

A distinction is made in the literature between the risk to an individual, the individual risk, and the risk to groups of people, known as societal risk. In both cases, the definition of risk is reduced to a point value, usually the mean risk.

Many definitions of individual risk exist. The definition used for the purpose of risk management policy in the Netherlands (Versteeg, 1992) is the following:

> "Individual risk is defined as the expected frequency of death due to a hazard of a hypothetical unprotected person, who is permanently located out of the doors, at any given fixed location beyond the perimeter of the installation concerned"

But the definition can be more general as well, such as the one provided by the Institute of Chemical Engineers (Ichem, 1985):

> "The individual risk is the frequency at which an individual may be expected to sustain a given level of harm from the realization of specified hazards."

In the U.S. nuclear risk management field, the individual fatality risk is further refined: the Quantitative Health Objectives make a distinction between individual early fatality risk (mainly an individual's probability of becoming a prompt casualty of a reactor accident in a given year) and the individual latent fatality risk, for which the death occurs many years later.

V.A.2. Societal risk

Parallel definitions exist for societal risk. The most widely used definition for societal risk is the one proposed by the Institute of Chemical Engineers (Ichem, 1985) which defines societal risk as:

> "The relationship between frequency and the number of people suffering from a specified level of harm in a given population from the realization of specified hazards". The definition does not give further precisions on what is meant by "harm".

The term "societal risk" has been traditionally associated with the number of fatalities in the case of an accident. However, others have seen societal risk as a much broader concept, including fatalities as well as other aspects of harm.

Experts have proposed (Ball and Floyd, 1998) to distinguish four categories of societal risks: the "collective risks", the "simple societal risks", the "diverse societal risks", and the "societal concerns". These categories are not mutually exclusive but correspond to a progression in the definition of societal risk and in the complexity of the tools to assess it, from the easily defined collective risks to the highly political "societal concerns".

- The first category (*collective risks*) deals with the diffuse risks associated with normal activities, such as radiation from nuclear materials or waste during normal activity. Generally, this type of risk is dealt with by setting an individual limit and by using cost-benefit analysis methods. The risk to society can be expressed as the product of the individual risk by the total number of people exposed.
- The second category (*'simple' societal risks*) casts risk in term of the number of fatalities that could be caused by an accident. It is based on the principle that often, fatalities are the best surrogate to express the seriousness of an accident and provide a simplified basis for risk evaluation. The most common tools to assess the 'simple' societal risk are FN curves. Those curves will be defined in part VI.A. An example is shown on the following figure:

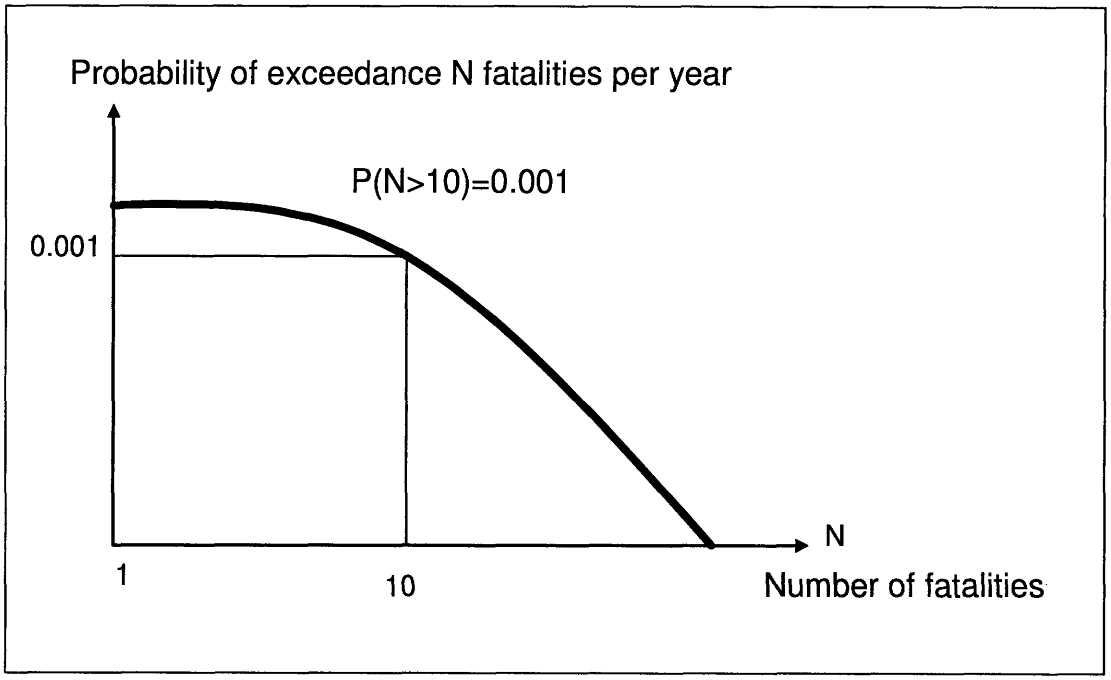

Figure 15: Example of FN curve, which displays the probability of having N or more fatalities per year, as a function of N, on a double logarithmic scale

- The third category[1] (*'diverse' societal risks*) consists in an extension of the second category and considers FN criteria based on fatalities but also on other types of harm such as environmental damage or injuries. However, few attempts have been made to translate such risk measures into criteria.
- Finally the fourth category (*societal concerns*) deals with risks from both normal activity and accidents and is the most complex one. The authors insist that:

  "At the policy or decision-making level, the crude use of FN curves based on fatalities is meaningless [...]. Nuclear FN curves tend to have a long tail at low probabilities, but it is seldom made clear that it is predicted on fundamental assumptions about the shape of the dose-response curve for ionizing radiation at very low doses, itself an increasingly hotly-debated topic [...]. Clearly, at this strategic level, decision-makers have to consider the full range of potential impacts (associated with both 'normal' activities and accidents) including fatalities, non-fatal injuries, property damage, environmental impacts, psycho-social harm, economic loss, business interruption costs, and even the political consequences of major accidents. Open-ended definitions of this kind tend to be anathema to those whose focus is numerical analysis since many of the components are difficult to handle if not beyond quantification and, even if quantifiable, could not be easily assimilated into a

[1] Ball and Floyd regroup the second and the third categories into a wider one called "societal risks".

decision model. In practice, however, optimum decisions can ultimately only be made by considering all of the goals and all of the consequences of various decision options, and in the final stages of policy formulation it is imperative that this be done".

On that level, attempts to quantify societal risk are currently being made. The ExternE project (Hirschberg, 1999) for instance aims at quantifying the external costs (production and transportation costs) of different sources of electricity: wind, solar, nuclear, biomass, coal, oil, natural gas, and hydroelectric. This interdisciplinary project uses Life Cycle Analysis (LCA), a method used to identify in details the inventory of material and energy flows associated with all stages of the life of an activity. The idea is that external costs of electricity have to be understood and known in order to be able to internalize the cost in the price of electricity, and make more rational energy choices. For instance, the occupational risk of coal miners or the environmental cost of pollution should be included in the price of coal. As for nuclear, the study aims at quantifying the external costs of the entire fuel cycle, including societal risk due to routine operation and accidents of nuclear power plants.

Table 4 summarizes the different categories determined by Ball and Floyd:

	Risk associated with:			
	'Normal activity'	Accidents	Suggested term	Type of criteria
Diffuse risk associated with exposure to hazardous material	Yes	No	Collective risks	Individual risk + Cost-Benefit Analysis
'Simple' risk associated with hazardous installations/activities which can be easily compared	No	Yes	Societal risks	FN criteria based on fatalities
'Diverse' risks associated with hazardous	No	Yes	Societal risks	FN criteria based on

installations/activities which required a broader basis for meaningful comparison				fatalities and other types of harm
Comparison of overall impacts/risks of technologies/strategies	Yes	Yes	Societal concerns	Political judgment - possibly aided by multi-criteria 'techniques'

Table 4: Categories of risks
(Ball and Floyd, 1998)

V.A.3. Why isn't putting a limit on individual risk enough?

Individual and societal risks deal with different issues. Putting a limit on individual risk is an equity measure, meaning that each individual is entitled the same level of safety. Most countries that have chosen to put a quantitative limit on risk have included a limit on individual risk, be it the United States, the Netherlands or the United Kingdom.

Individual risk does not take into account the total number of people exposed to the hazard. On the contrary, societal risk is a function of the total population exposed. Two nuclear power plants, each complying with the QHOs, can entail different societal risks.

Let's develop a very simple example in order to illustrate this fact:

- Consider two identical nuclear power plants, one for which there are 1000 people located in the vicinity of the plant, the other one for which there are 100,000 persons. Assume furthermore that two independent scenarios only can lead to fatalities: 'Scenario A' has a likelihood of 10^{-4} per year and 'Scenario B' a likelihood of 10^{-5} per year. Assume also that, if 'Scenario A' occurs, one person out of five located in the vicinity of the plant will die, and one out of two for the same region but considering 'Scenario B'.
- The *individual fatality risk* is equal to $\frac{10^{-4}}{5}+\frac{10^{-5}}{2}=2.5*10^{-5}$ per year for both plants.

- There are several ways to calculate the *societal risk*, depending on its definition. If we assume the societal risk is captured by the expected number of fatalities per year, the societal risk in the first case is $2.5*10^{-5}*1000 = 2.5*10^{-2}$ expected fatalities per year, whereas it is equal to 2.5 in the second case.

This example uses simplistic assumptions. In general, there is no direct relation between individual and societal risk. However, it illustrates the fact that two systems complying with individual risk limits might have different societal risk profiles.

Figure 16 illustrates the different parameters considered:

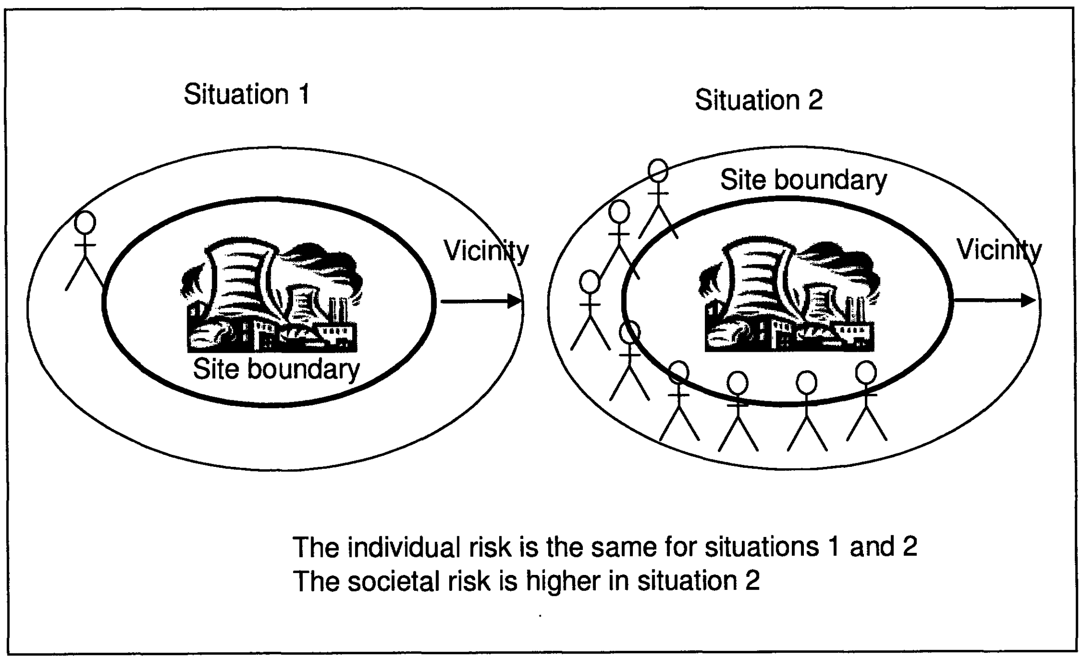

Figure 16: Illustration of the difference between individual and societal risk

V.B. General questions on societal risk

V.B.1. At what level societal risk should be considered?

The Ichem definition of societal risk does not specify the level (e.g. site, regional, national) at which societal risk should be considered.

The approach chosen by the Dutch Ministry of Housing, Land Use Planning and Environment (VROM) assumes that societal risk is measured at the level of an installation (Versteeg, 1992).

However, other "levels" of societal risk are possible. Vrijling argues that certain risks that seem acceptable at an individual/site level may not be acceptable at the national level (Vrijling et al, 1995). He concludes that acceptance of societal risk takes place at a national level. A convincing example that he chooses to present is the commercialization of a toy that would cause a child's death at a frequency of 10^{-4} per year per toy. Compared to risk from other accidents, one can say that from an individual point of view, the toy is relatively safe. If 1000 children use the toy, the expected number of death is 0 or 1 per year. However, if the toy becomes very popular and 10^7 toys are sold, the expected number of deaths will be 1000, which is clearly unacceptable at a national level.

It is interesting to ask if the previous example applies to nuclear power plants. The idea in the toy example is that the toys are identical, and therefore the risks perfectly correlated. This is not true for nuclear power plants. On a high-level, two plants can differ by their design and the way they are operated (for instance the safety culture might be different). In the United States, there is a variety of vendors and utilities, which favors a site-level approach. This might not be valid in other countries: for instance in France, all reactors have been built by the same vendor, and there is currently only one utility, Electricité de France, operating them.

In the United Kingdom, the Health and Safety Executive (HSE, agency responsible for health and safety regulations in Great Britain) has argued that since any plant in the country could be the source of an accident, it is not the risk per plant that matters but the risk attached to the whole family of plants (HSE, 1992). On the other side, imposing a national limit on societal risk would limit the total number of reactors that could be built.

Defining the level at which societal risk should be considered is a matter of policy. The 1986 individual risk limits were not intended to be used at a site level. However, in practice, they were used as benchmarks for individual plants. Therefore, a site-level is more appropriate for nuclear power plants in the U.S.

V.B.2. Should risk aversion be included in the societal risk measures?

V.B.2.a. Definition

Risk aversion is a concept explaining the behavior of consumers and investors under uncertainty. It is generally defined as the reluctance of a person to accept a lottery with an uncertain payoff rather than another lottery with more certain but possibly lower expected payoffs. In the context of societal risk, a key question is whether more weight should be given to a large accident with many fatalities than to several smaller accidents producing the same total number of fatalities. We will therefore refer to this concept as the multiple-fatality aversion concept. Three attitudes towards multiple fatalities are possible: risk neutrality, risk aversion, and risk proneness, as illustrated on Figure 17.

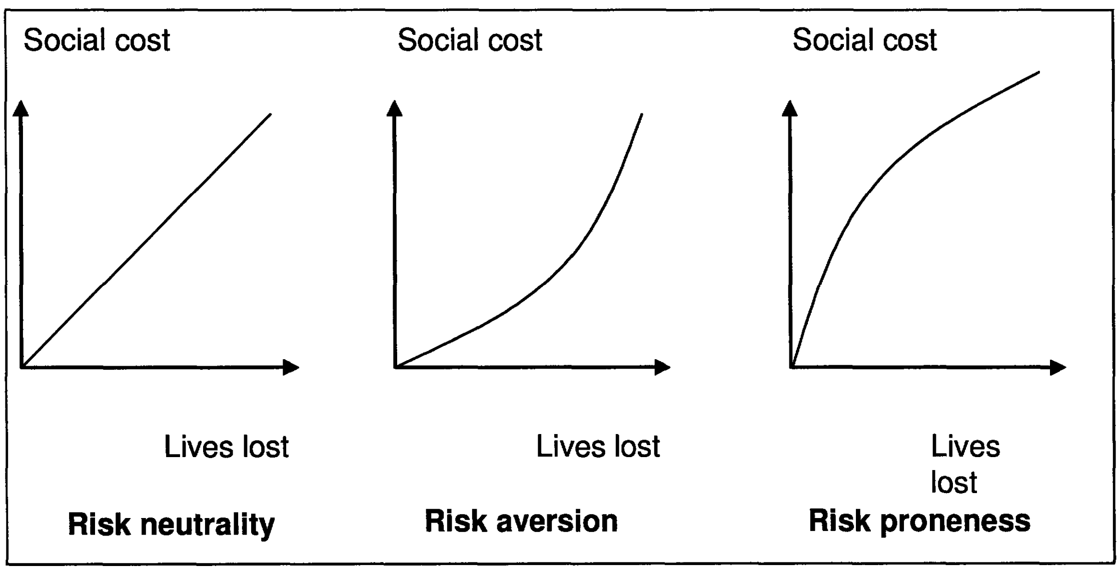

Figure 17: Categories of multiple-fatality risk aversion

(Slovic, 1984)

Policy makers have generally relied on the idea that society was multiple-fatality averse, an assumption revisited by Slovic (Slovic, 1984). According to Slovic, modeling the impact of an accident by a risk-averse function of the number of fatalities is inadequate. Accidents with very few fatalities or even none may have higher societal costs then accidents involving more fatalities. For him, accidents serve as "signals" of the nature and controllability of risk they imply. For instance, an accident whose consequences are well understood, familiar and with little potential for a catastrophe will have a much lower social cost as an accident with the same number of fatalities but that does not meet the previously mentioned criteria.

Keeney (Keeney, 1980) has presented several assumptions that he has proved to lead to risk-proneness: the first assumption states that a sure loss of N persons is less desirable than the 50-50% chance of losing either 2N or 0 person(s), which was supported by some empirical evidence. Second, Keeney argues that as N gets larger, each incremental life lost has less marginal societal impact.

Ball and Floyd of the British Health and Safety Executive support a risk-neutral position:

> "Though documented evidence is sparse, nowhere have we found any compelling support of arguments for an ex-ante stance of other than risk-neutrality in societal decision making" (Ball and Floyd, 1998).

V.B.2.b. Inclusion of risk-aversion into criterion and measure

There are several ways to model risk-aversion:

- The most commonly used method is to include a risk aversion factor: for instance the societal cost C(N) of N fatalities can be taken to be equal to N^{α}, with $\alpha > 1$:

  $C(N) = N^{\alpha}$ (Equation 1)

  The societal cost of having 2N fatalities is more than twice costlier as the societal cost of having N fatalities since C(2N) $=(2N)^{\alpha} = 2^{\alpha} N^{\alpha} > 2(N^{\alpha})$. This has been referred to as the α-model by Slovic (Slovic, 1984) and has been applied in the Netherlands with a factor equal to 2.
- Another possibility consists in integrating the standard deviation into the equation. A measure of total risk (TR) defined by the sum of the expected value of the number of fatalities and the standard deviation multiplied by a risk aversion factor k has been proposed (Jonkman et al, 2003): $TR = E(N) + k\sigma(N)$ (Equation 2)

It is of course possible to derive risk-neutral societal risk criteria by setting the risk aversion factors respectively equal to 1 for α in Equation 1 (this approach has been chosen in the United Kingdom and is known as the Canvey line, but it is not used to assess the tolerability of nuclear risk) and 0 for k in Equation 2.

V.C. Quantitative risk limits in the United States

V.C.1. The 1986 Safety Goals

The process of defining quantitative risk limits in the United States was a long and complex one. In 1986, the U.S. Nuclear Regulatory Commission adopted a Policy Statement on Safety Goals for nuclear power reactors (USNRC, 1986) in order to define an acceptable level of radiological risk, and stated that there were two qualitative safety goals:

> "Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health
> Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks."

The Commission translated these qualitative safety goals into quantitative ones; known as the Quantitative Health Objectives (QHOs) and previously cited in Part III. We remind here their statement:

> "The risk to an average individual in the vicinity of a nuclear power plant (region between the site boundary of the power plant and one mile beyond this boundary) of prompt fatalities that might result from reactor accidents should not exceed one tenth of one percent of the sum of prompt fatality risks that result from other accidents to which the U.S. population is generally exposed.
> The risk to the population in the area near a nuclear power plant (region between the site boundary of the power plant and ten miles beyond this boundary) 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 risks from all other causes."

Many have argued that the quantitative safety goals did not take into account the total societal risk by imposing a limit on the total number of fatalities that could result from a nuclear accident.

This issue was addressed as soon as 1986, in the Safety Goals Policy Statement itself, by Commissioner Bernthal in his separate view on Safety Goals Policy:

> "As they stand, these 0.1 percent goals do not explicitly include population density considerations; a power plant could be located in Central Park and still meet the Commission's quantitative offsite release standard" (USNRC, 1986).

The issue was raised periodically afterwards as a modification of the Safety Goals was prepared. However, this modification was never achieved and it seems today that the very same Quantitative Health Objectives will be used for the next generation of reactors (USNRC, 2006).

V.C.2. Surrogate Risk Metrics

Showing compliance with the QHOs requires a level-3 PRA that calculates the risks to an individual. Those PRAs require an intense modeling of the event sequences, and the uncertainties are very high. Those uncertainties exist independently of the PRA (for instance the health effects due to radiation exposure are uncertain), but the PRA displays these uncertainties, and deciding on the acceptability of risk might be difficult.

To deal with this issue, surrogate risk metrics were developed. A level-1 PRA is necessary to calculate the CDF of a plant. A level-2 PRA calculates both the CDF and the LERF. For current reactors, the limits were set at 10^{-4} /reactor year for the CDF and 10^{-5} /reactor year for the LERF. Interestingly enough, the limits are put on the frequency, no matter of the possible consequences (for instance, the consequence of a large early release depends on the core inventory).

V.D. Sources of societal risk

If we consider a nuclear power plant, there are mainly three sources of societal risk increase: degradation of the plant safety, increase of the core inventory, or an increase in the number of people around the plant. These two issues are very different. The first deals with plant characteristics, the second with siting decisions.

V.D.1. Issue of siting

Societal risk criteria are closely related to siting decisions. Indeed, one of the arguments for not defining a societal risk limit in the United States is that it is already included in 10 CFR Part 100.

Several points have to be made regarding 10 CFR Part 100:

- *No allowable population density around a reactor is quantitatively specified.* Section 3 provides a definition for the Low Population Zone (LPZ), which is:

  "The area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case." Section 10 only mentions the population density as one of the factors that should be considered for evaluating a site: "Population density and use characteristics of the site environs, including the exclusion area, low population zone, and population center distance"

- *An accident is postulated to assess the acceptability of a site but the quantitative dose limits only apply to individuals.* Indeed, section 11 states that:

  "As an aid in evaluating a proposed site, an applicant should assume a fission product release from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:

  - *An exclusion area* of such size that <u>an individual</u> located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.
  - *A low population zone* of such size that <u>an individual</u> located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.
  - A *population center distance* of at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, the boundary of the population center shall be determined <u>upon consideration of population distribution.</u> Political boundaries are not controlling in the application of this guide. Where very large cities are involved, <u>a greater distance may be necessary</u> because of total integrated population dose consideration."

Note that the definition of the *population center distance* refers to the population density around the reactor but gives no quantitative indications.

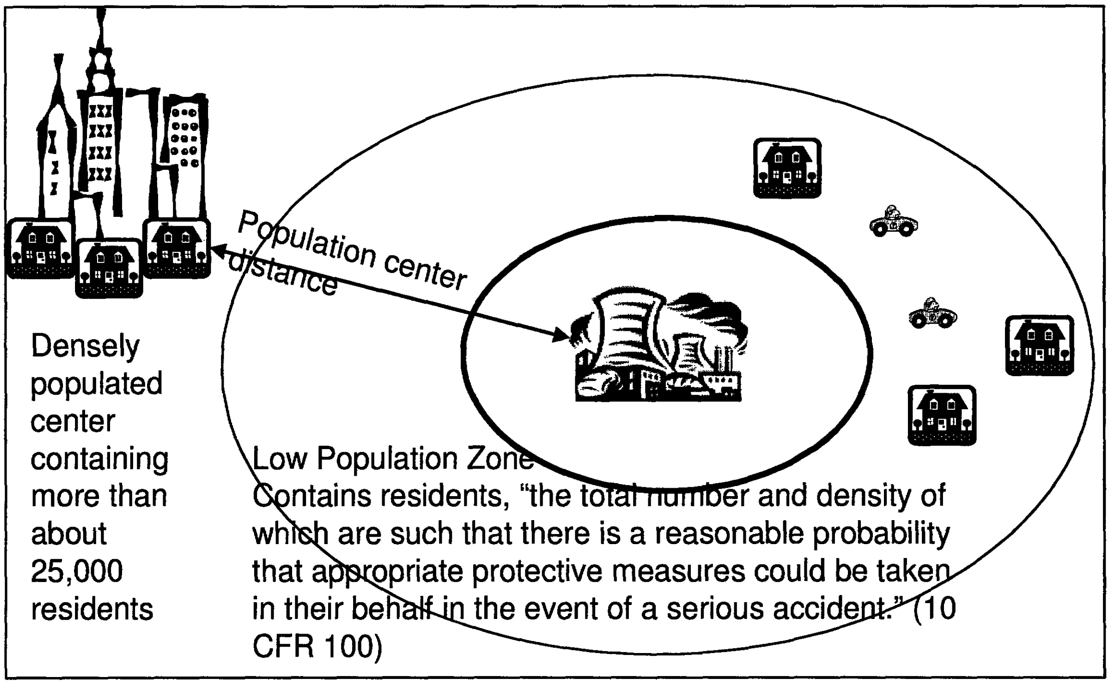

Figure 18: Siting distances in 10 CFR 100

There is a trade-off between safety and siting convenience. The closer a plant is from a metropolitan area, the lower are the costs but the higher are the societal risks. One can also remind briefly the Ravenswood siting controversy, when a nuclear power plant was proposed for construction in the highly populated Queens borough of New York City.

To conclude, one can say that 10 CFR Part 100 deals with societal risk in a qualitative manner but fails to define an acceptable quantitative level of societal risk.

V.D.2. Plant characteristics

Even if there is no change in the population around the plant, societal risk may increase due to plant characteristics themselves:

- If the probabilities of event sequences leading to an accident increase: for instance, for a given number of people living around the plant, if the probability of an accident sequence goes up because of the aging of certain components, societal risk is higher.
- If the potential consequences of event sequences increase: for instance if more radioactive material can be released during an accident. This is the case when utilities are granted power uprates. Indeed, highly enriched uranium is generally added, which increases the total core inventory. In turn, societal risk goes up.

Utilities submit power uprates as license amendment requests. It must be proved that the plant will remain safe, and that there are still adequate measures taken to protect the health and safety of the public. However, the increase in societal risk is not taken into account quantitatively. Power uprates are widely used by the industry: in July 2004, the USNRC had completed 101 power uprate reviews, resulting in a gain of approximately 4,000MWe.

V.E. Conclusion

At a national level, measuring societal risk and assessing its tolerability is highly complex. Indeed, using nuclear technology entails both direct benefits (e.g. available energy) as well as unquantifiable positive externalities (e.g. energy independence). There is no easy way to weight these benefits against the existence of very low probability and high consequence events, able to kill many. Risks at that level have been referred to as "societal concerns", and no quantitative tools can easily help to decision-making. However, if we restrict the analysis to the risk for people around a power plant, such tools exist and are already in use in countries such as the Netherlands. However, one must keep in mind that part of the difficulty in implementing societal risk requirements comes from the tremendous role played by public perception and its reluctance to accept the possibility, even with very low probability, of high consequence events.

Part VI. Overview of Quantitative tools to measure societal risk

Limiting societal risk is a complex issue. The USNRC has recently considered the use of F-C curves, but on an individual event basis only. No goal limiting the societal consequences of nuclear accidents and operation is included in the framework so far. Including such goal, as it is done in the Netherlands by putting a limit on the total number of fatalities resulting from a potential accident, is a possibility.

The risk curves that have been used for societal risk assessment have mostly referred to one type of consequence, usually the total number of fatalities. These curves are called FN curves. Extended measures have been proposed, but have never entered regulations. An overview of different quantitative tools to assess societal risk is presented in the following paragraphs.

VI.A. FN curves

VI.A.1. Definition

An overview of quantitative risk measures of societal risk is provided in (Jonkman et al, 2003), in which societal risk is assumed to be related to the number of fatalities. Among them are FN curves, which display the probability of having N or more fatalities per year, as a function of N, on a double logarithmic scale.

We have $1-F_N(x) = P(N \geq x) = \int_x^{\infty} f_N(n)dn$, where f_N is the probability density function of the number of fatalities per year, $F_N(x)$ is the cumulative distribution function of the number of fatalities per year, and $1-F_N(x)$ is the complementary cumulative distribution function (probability of having x or more fatalities per year).

Figure 19 presents FN curves for different groups of activities in the Netherlands:

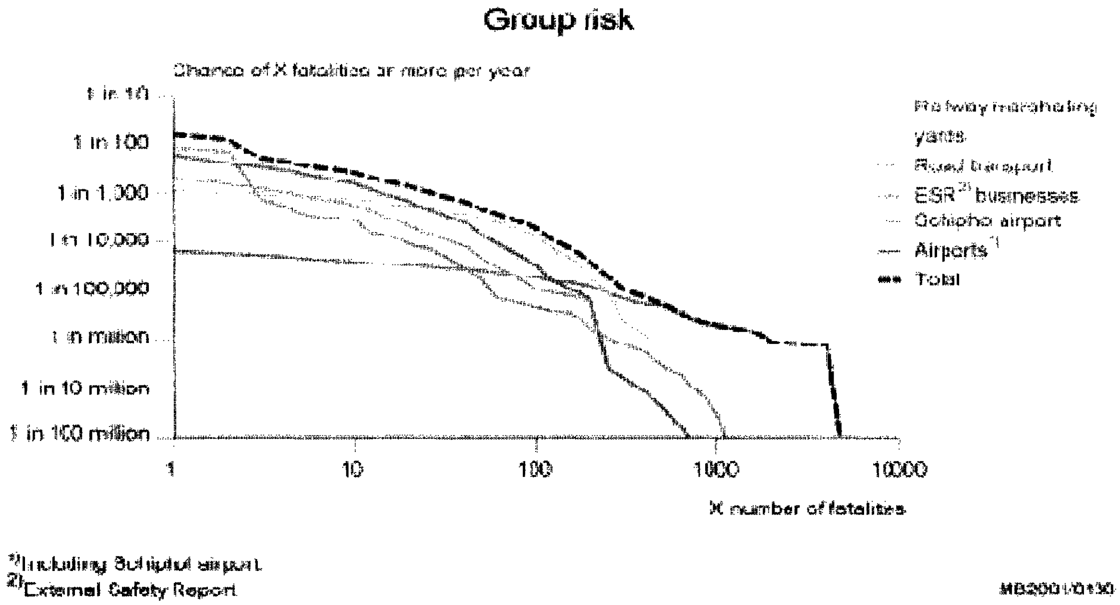

Figure 19: Example of FN curves for different groups of activities in the Netherlands

(Source RIVM, 2001)

VI.A.2. Different types of FN curves

A distinction must be made between:

- The FN curves that display historical records of accidents. They are built using historical data;
- The FN curves that result from quantitative risk assessment. Those are the result of modeling. For instance, a level 3 PRA would be needed in order to build such curve for a nuclear power plant. In practice, FN curves are often a mix of historical / empirical data and modeling (Evans, 2003).
- Finally, the FN criteria are the curves that are used to assess the tolerability of FN curves.

VI.A.3. Use of FN curves in the Netherlands

FN curves are in use in some European countries such as the Netherlands and the United Kingdom, for the purpose of societal risk management.

The Netherlands is a small country, with an area slightly less than twice the size of New Jersey, and a total population of around 16.5 million as of 2006 (CIA data, 2006). Compared to the United States, the number of inhabitants per square kilometer is 15 times higher in the Netherlands. Lack of space is a significant issue in this country, which could account partly for their decision to use societal risk criteria. Major accidents in the 70's involving Liquefied Petroleum Gas (LPG) stations focused the attention on risk assessment and reduction, and on the need for national standards (Ale, 2005). The first document to introduce limit values for individual and societal risk was issued in 1986 and focused on LPG accidents. The policy framework was then integrated in the document "Dealing with risk" that accompanied the First National Environmental Policy Plan in 1989. The individual and societal limits set were also to be used for nuclear power plant policy.

In the Netherlands, probabilistic safety criteria and goals have been developed. The risk management policy (Versteeg, 1992) adopted for potential hazardous industries explicitly refers to the safety of each single individual in the vicinity of the plant and to the population as a whole and consists of different steps. The first step consists of the identification of the hazards and risks and the scenarios that lead to then. These scenarios are then quantified with probabilistic risk assessments methods. A third step, called the "assessment step" consists of showing compliance with criteria and objectives. Risk is reduced until an optimum level is reached, following the As Low as Reasonably Practical (ALARP) principle. Finally, control is implemented to ensure that risk is maintained at this optimum level.

The policy uses a three-region approach and distinguishes three risk-related regions: one where acceptable activities lie, one where reduction of risk is necessary according to the ALARP principle and a last region where risk is considered unacceptable. The first separation is a de minimis value, the second the criterion itself, which is usually referred to as the VROM criterion.

For instance, for each source of activity, the upper bound of acceptable individual level of risk is 10^{-6}/year, while the de minimis value is 10^{-8}. Between those two values, the ALARP principle is applicable. For all hazardous sources or activities, the maximum acceptable level of risk is 10^{-5}.

Regarding societal risk, a curve relating the exceedance frequency of N or more fatalities to the number of fatalities is used at the plant level. No national societal goal has been proposed. Two complementary cumulative density functions are used to determine the three regions. The lines chosen are two straight lines with a slope -2 reflecting risk aversion.

The following figure gives a visualization of the FN criterion adopted in the Netherlands:

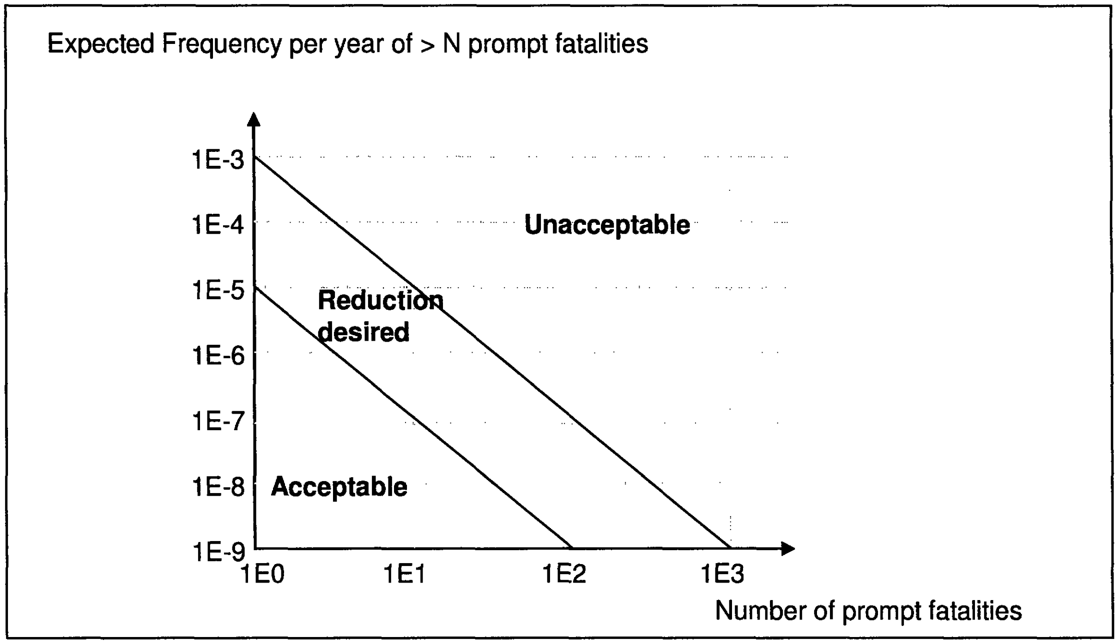

Figure 20: FN criteria in the Netherlands (note that only the upper curve is a criterion)

(Versteeg, 1992)

These criteria are on the number of fatalities outside the side boundary, therefore apply to nuclear major accidents and do not apply to workers.

It is possible to compare the Dutch criterion with the Canvey Line criterion, defined by the British Health and Safety Executive when it assessed in a milestone study in 1978-1981 the potential of the industrial installations at Canvey Island on the Thames for causing a major accident affecting the surrounding population (Ball and Floyd, 1998). The comparison of the two criteria is presented on Figure 21:

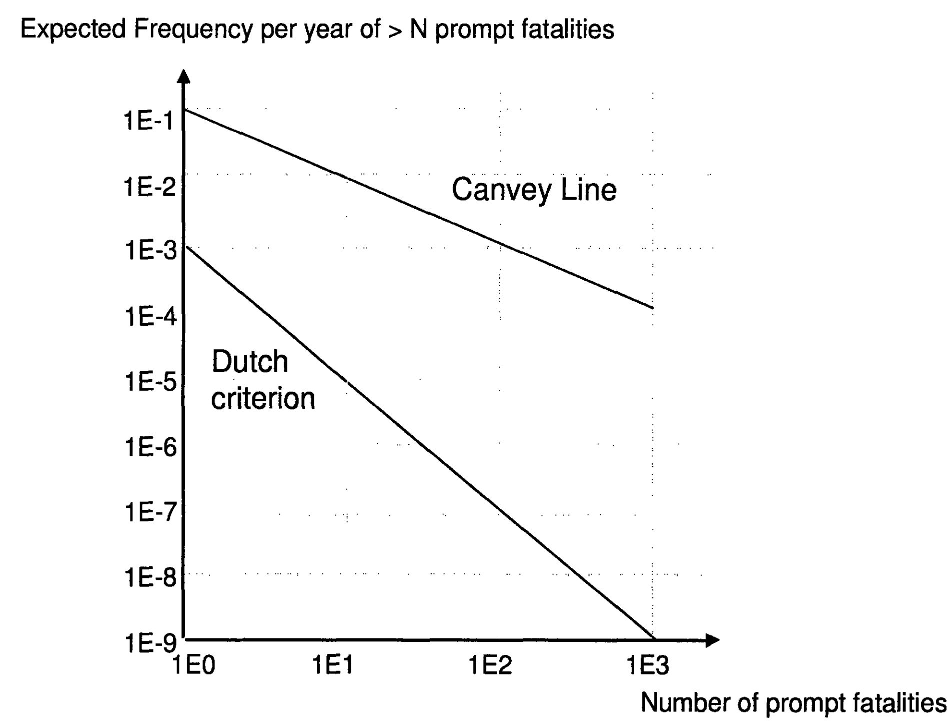

Figure 21: Comparison between Dutch and British risk tolerability criteria (the Canvey Line criterion is risk-neutral, as opposed to the highly risk-averse Dutch criterion.)

VI.A.4. Dutch regulations and U.S. PRA results

Dutch FN curves are highly risk averse. Indeed, if we assess the tolerability of risk of certain U.S. plants using the Dutch criterion, the results might be surprising: Figure 22 shows a risk curve (which is similar to a FN curve) for the total number of early fatalities at a nuclear power plant from NUREG-1150 results (USNRC, 1990). The Dutch criterion is superimposed on the figure.

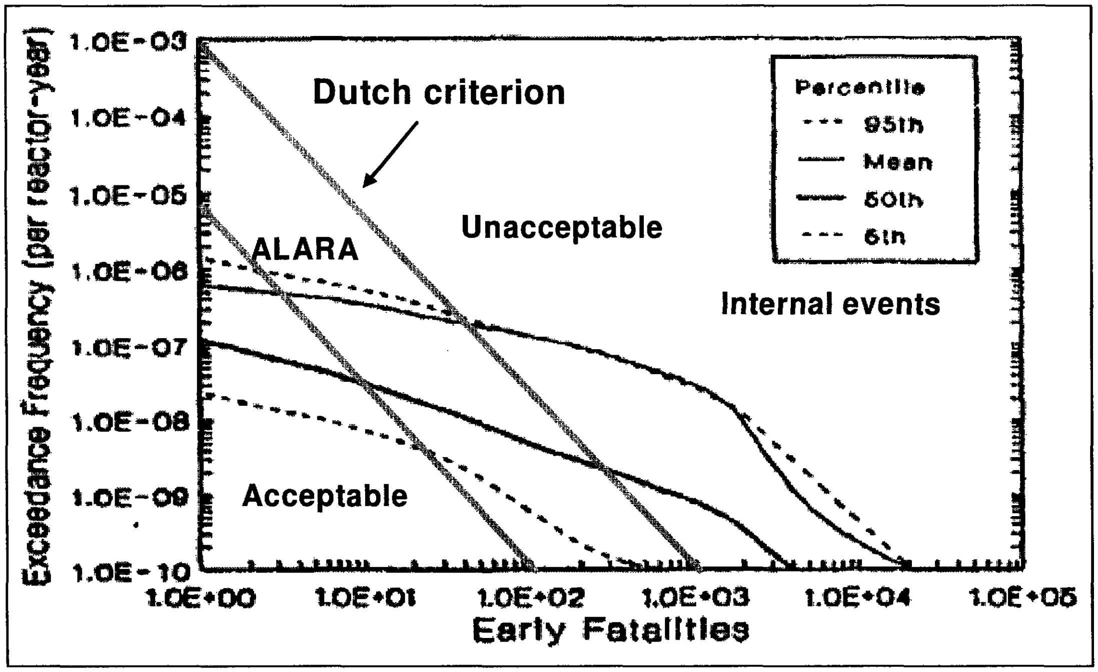

Figure 22: Example of level-3 PRA results (NUREG-1150) and Dutch criterion

There are regions where the mean value risk curve is above the criterion, which is unacceptable in the Dutch view. Furthermore, the assumptions between the Dutch and U.S. approaches are very different. Indeed, such risk curve for NUREG-1150 was obtained making the assumption that 99.5% of the population was evacuated; this assumption is not made in the Netherlands:

> "In demonstrating compliance with the risk criteria, it is necessary to assume that only the usual forms of mitigating measures are taken (i.e. action by fire services, hospitals, etc.). Although special measures like evacuation, iodine prophylaxis and sheltering may be taken by the Emergency Preparedness Organization, these are disregarded in the analysis. In the Dutch view, it is unreasonable to assume that any countermeasure will be 100% effective. On the contrary, it is more realistic to expect that a substantial part of the population will be unable or unwilling to adopt the prescribed countermeasure. The PSA results used to demonstrate compliance with the risk criteria need, therefore, to reflect this more conservative assumption. However, for the sake of interest, the PSA results of the Dutch nuclear power plants show both situations: with and without credit being given for countermeasures." (VROM, 2005)

However, there is today only one nuclear reactor in the Netherlands (PWR, 452 MWe).

If we don't consider the different assumptions, we can see that the result would have been very different with a less conservative criterion:

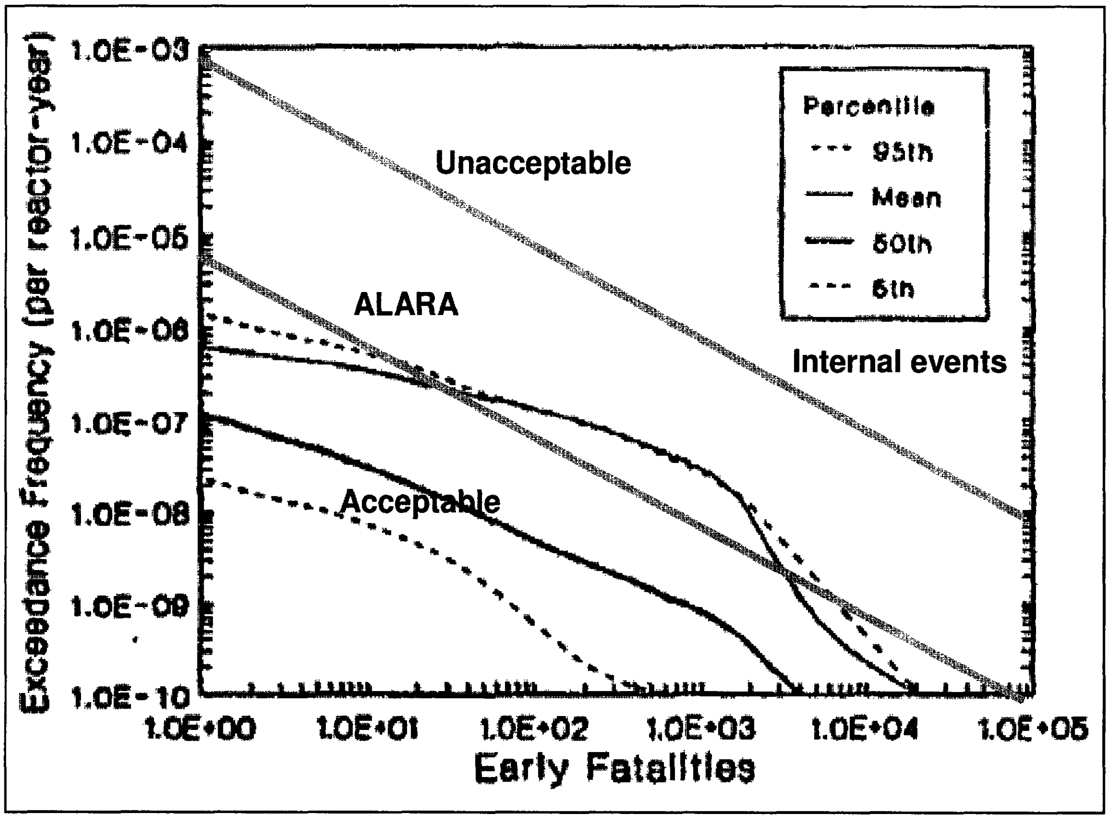

Figure 23: Example of level-3 PRA results (NUREG-1150) and risk-neutral criterion

With a slope equal to -1 and assuming that 99.5% of the population evacuates, the mean FN curve lies below the criterion. This is also the case when the slope of the criterion line is set at 1.2, as illustrated on Figure 24:

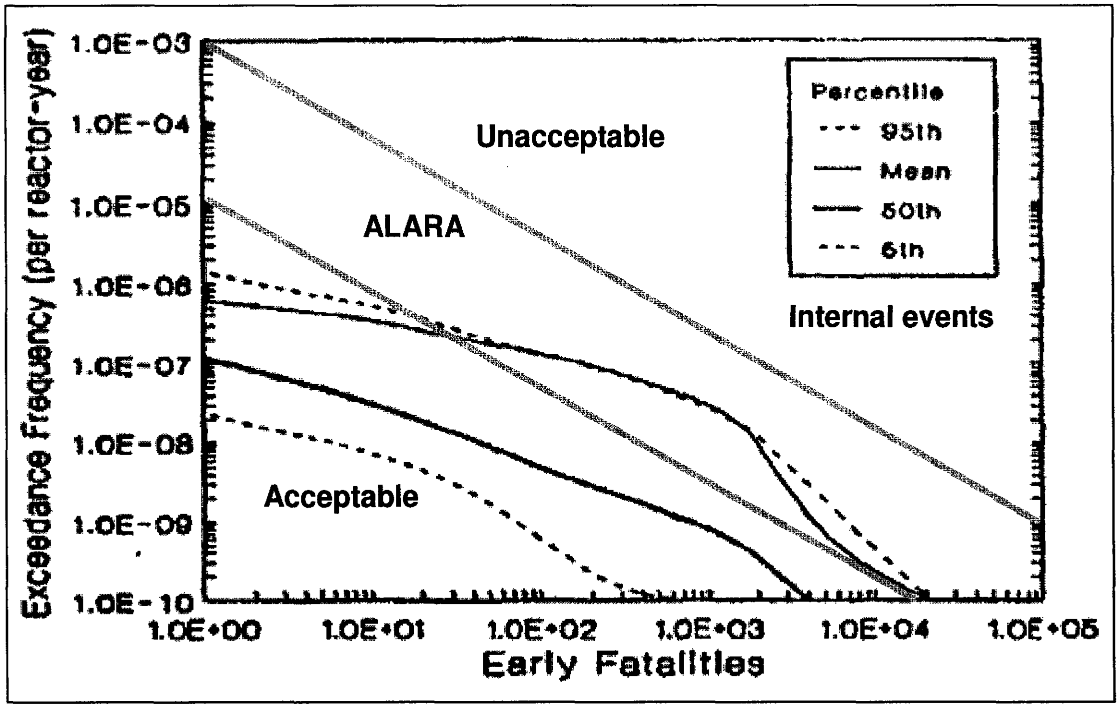

Figure 24: Example of level-3 PRA results (NUREG-1150) and risk-averse criterion (slope equal to 1.2)

VI.A.5. Limits of FN curves

The use of FN curves as a decision-making tool has been criticized for the following reasons:

- FN curves correspond to a minimax decision rule. Therefore, they concentrate on extreme features of statistical distribution, which can lead to decisions that appear unreasonable (Evans and Verlander, 1997)
- As opposed to expected disutility functions, FN curves lead to "incoherent" judgment, in the language of decision theory, when there is uncertainty associated with the accidents (Evans and Verlander, 1997)
- FN curves are based solely on the number of fatalities. Decisions that use FN curves as a risk-assessment tool overlook important consequences of accidents.

VI.B. Other risk assessment measures

Other societal risk measures have been proposed for use, in an attempt to solve the issues associated with FN curves previously identified.

For instance, FN curves do not allow the comparison of different systems, which could be possible if a system could be represented by a single value, and not a curve. A simple measure of societal risk can be expressed by the expected number of fatalities per year (Ale et al, 1996), which is equal to the integral of the FN curve (Vrijling et al, 1997).

$$E(N) = \int_0^\infty x f_N(x)dx = \int_0^\infty (1 - F_N(u))du$$

Other measures also exist, that take into account risk-aversion. The aversion is taken into account through a coefficient that gives more weight to accidents with a large number of fatalities. For instance, the British Health and Safety Executive (HSE) defined a weighted risk integral parameter called the Risk Integral as (Jonkman et al, 2003):

$$RI_{COMAH} = \int_0^\infty x^\alpha f_N(x)dx$$

Evans and Verlander propose another measure of societal risk: the expected disutility (Evans and Verlander, 1997). According to the theory of decision-making under uncertainty, the tolerability decisions must be made on the basis of expected utility to be consistent. The first step is to associate a number u(n) as a measure of harm (u increases with n and has the same properties as a utility function). It is assumed that the disutility function satisfies the axioms of the Expected Utility Theory.

The disutility of an accident of uncertain size in the engineering system is given by:

$$u_\alpha = \sum_n u(n)p(n)$$

The choice of the disutility function can reflect risk-aversion (for instance, $u(n) = n^\beta$ with $\beta > 1$).

Very few societal risk measures allow the consideration of consequences other than fatalities. This possibility should thus be explored.

VI.C. Extended measures of societal risk

VI.C.1. Would an extended definition of societal risk be more appropriate?

An extended definition of societal risk might be more appropriate depending on the technology whose risk is studied. For instance, the risk metric used to assess the societal consequences of car accidents in the U.S. measures the total number of prompt fatalities per year. It could include injuries as well. Societal consequences of dam failure include among others: prompt fatalities, evacuation costs, and off-site property damage. Each category of activity entails specific risks and hence, specific risk assessment tools. It is therefore necessary to investigate the societal consequences of nuclear accidents. A brief overview of Chernobyl and Three Mile Island accidents is provided in the following paragraphs.

VI.C.1.a. Three Mile Island

The Three Mile Island accident in 1979, that involved a partial core meltdown of one reactor, was the most serious nuclear incident in the United States commercial nuclear power plant operating history. Detailed studies were conducted to assess the radiological consequences of the accident by the USNRC, the Environmental Protection Agency, the Department of Health, the Department of Energy and the State of Pennsylvania, as well by independent groups. No adverse effects from radiation on human, animal and plant life could be directly correlated to the accident (USNRC factsheet). However, it is important to note that 12,000 people were asked to evacuate the area (families with pregnant women and preschool children living within 5 miles of the facility), and an estimated 144,000 persons within 15 miles evacuated for a period averaging between 4 and 5 days (Houts et al, 1988). Long term evacuation rates, i.e., people permanently moving out of the area, were not affected by the crisis. Short term costs were much lower than for natural disasters, because it involved no physical damage and consisted mainly of expenses borne by families who evacuated, and loss of sale and production costs for businesses. There was little evidence of the long-term economic impact on people living in the vicinity, for instance regarding real-estate.

During the crisis, there was an estimated 10% increase in the number of patients that reported symptoms characteristic of mental patients, but after 18 months, it was no longer higher than in the rest of the population studies.
The costs of cleaning up the damaged reactor were substantial. Public fear and distrust towards nuclear power greatly increased.

VI.C.1.b. Chernobyl

The Chernobyl accident occurred in Ukraine in 1986 and is the most serious nuclear accident in the history of commercial reactors worldwide. The consequences of the accident are still imperfectly determined. However, Dr. El Baradei, IAEA Director General, has classified them in three categories in a 2005 IAEA conference entitled "Chernobyl: Looking back to go forwards": the physical impacts, in terms of health and environmental impacts, the psycho-social impacts on the populations and the influence of the accident on the nuclear industry worldwide.

The following figures were cited in his speech:

- Among the emergency rescue workers at the scene of the accident, around 50 individuals died either from acute radiation syndrome in 1986 or due to other radiation-related illnesses in the year since.
- About 4000 children and adolescents contracted thyroid cancers from ingestion of contaminated milk and other foods, and 9 of those children have died.
- Overall, based on statistical modeling of the radiation doses received by workers and local residents, a total of 4000 deaths will eventually be attributable to the Chernobyl accident.
- Environmental fallout from the accident affected croplands, forests, rivers, fish and wildlife, and urban centers. In the three countries more affected, nearly 800,000 hectares of agricultural land was removed from service, and timber production was halted for nearly 700,000 hectares of forest.
- The psycho-social impacts were also devastating. Over 100,000 people were evacuated immediately after the accident, and the total number of evacuees from severely contaminated area eventually reached 350,000 people. While these resettlements helped to reduce the collective dose of radiation, it was deeply traumatic for those involved. Studies have found

that exposed population had anxiety levels twice as high as normal, with a greater incidence of depression and stress symptoms.

As it is summarized by G. Saji (Saji, 2003), "As experienced in the Chernobyl accident, the psychological consequences, as a category of health effects may well be the most significant at the present time."

VI.C.1.c. The number of fatalities does not adequately capture societal risk

What we can conclude from the review of these nuclear accidents, especially through the example of Three Mile Island, is that societal consequences, and therefore societal risk, certainly should capture more parameters than only fatalities, for instance psychological damage to the population and land contamination.

The Dutch experience supports this conclusion (VROM, 2006). The country is currently reconsidering its way of addressing societal risk. The fireworks disaster of Enschede in 2000 led to an intensification of external safety policy and ambitious objectives were set out in the Dutch Fourth National Environmental plan. Research is currently undertaken in order to improve the framework used to limit societal risk. A full report will be submitted to the Lower House of the Dutch Parliament in the summer 2006. One of the issues identified so far is the need to identify the potential societal disruption of any prospective disaster, including injuries, damage to people, actions taken by the emergency services and disaster response services.

VI.C.2. Societal risk measures accounting for more than fatalities

VI.C.2.a. Swiss proposal of risk measure

Literature on possible "extended" measures of societal risk is scarce. In order to quantify the integrated impact of a scenario, there are mainly two possibilities: translate all consequences into monetary values or transform all consequences into no-unit values

Such approach was proposed in Switzerland. The federal ordinance on Protection against Major Accidents (BUWAL, 1991) was issued in April 1991 in Switzerland with the objective of protecting the public and the environment from major accidents. A new risk appraisal measure was proposed, which used a F-C curve, with the consequence being the aggregate measure of 9 parameters summarized in the following table:

		Indicator	**Description**
Impact on man, animals and ecosystem	N1 (Persons)	Number of fatalities (persons)	Early and latent fatalities
	N2 (Persons)	Number of injured	Serious and superficial injuries
	N3 (Persons)	Number of evacuees	Persons evacuated for more than a year
	N4 (Persons)	Alarm factor	[Duration of stress × number of affected people]
	N5 (Animals)	Number of dead animals	Big animals. Small animals count for 1/100 Fish belongs to next category
Impact on natural resources	N6 (sq. meters)	Area of damaged ecosystem	
	N7 (sq. meters)	Area of contaminated soil	Area that is no longer usable or inhabitable or that requires very expensive decontamination treatment
	N8 (sq. meters)	Area of polluted groundwater	
Impact on property	N9 (Swiss Francs)	Expenditures	Property damage, evacuation costs...

Table 5: Categories defined in Swiss proposal
(Buwal, 1991)

The proposed regulation assumes that damage can be represented by these 9 parameters. Fewer parameters might be selected, depending on the field of study. It is important to note that the selection of the categories is subjective.

A quantitative risk analysis is done, and each scenario is assessed in terms of its impact on the nine categories, as well as its frequency. Once the overall impact value of a scenario has been determined from the individual impacts on each of the categories, the scenarios can be ordered in terms of their consequences and complementary cumulative density function can be built. The proposed CCDF expresses the probability of exceeding a certain consequence per site and per year as a function of the overall impact.

A crucial question is how the different impacts should be combined in order to retain only one consequence value for each scenario. A suitable impact scale is defined for each category of indicator value, and then these individual impact values are combined to obtain the overall impact value. Hence, the extended damage assessment asks two main questions:

- How can individual indicators be appraised?
- How can the individual indicators be combined into a single consequence value?

A methodology is presented in (Bohnenblust et al, 1994) that uses the Fuzzy Set theory.

If different scales are available in literature to define the significance of an event (Bohnenblust et al, 1994), the Fuzzy Set theory uses a scale that ranges from 0 (normal operation) to 1 (catastrophe). If the impact value is over 1, it is then taken equal to 1. Figure 25 presents such disaster scale:

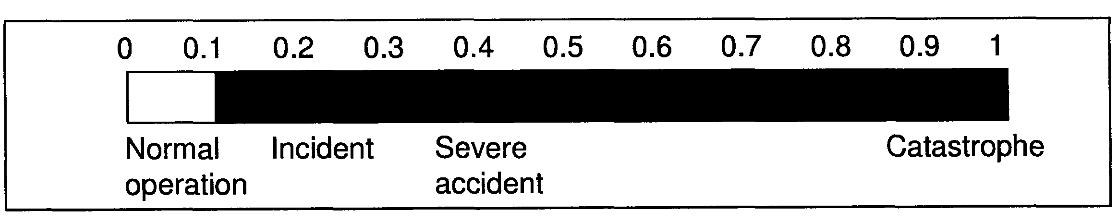

Figure 25: Disaster scale

(Bohenblust et al, 1994)

The Fuzzy Set theory uses membership functions, which are functions that define how each point in the input space is mapped to a membership value (or degree of membership) between 0 and 1. In our case, if the function equals one for an element x, then x necessarily possesses a predefined property. If the function equals zero, then it unequivocally does not possess the property. Finally, an intermediate value indicates the degree of membership or the degree to which x possesses the property.

Bohnenblust postulates a simple linear relationship between the logarithmic indicator value and the impact value. The same approach can be found in the Swiss Ordinance. To determine the function, the magnitude of the impact value 0.2 and 0.6 for each indicator is assessed subjectively by experts. The functions are noted $g_A^i(N_i)$

Figure 26 shows the relationship for fatalities (a total number of 4 fatalities is assigned an impact value of 0.2, 100 fatalities correspond to a value 0.6):

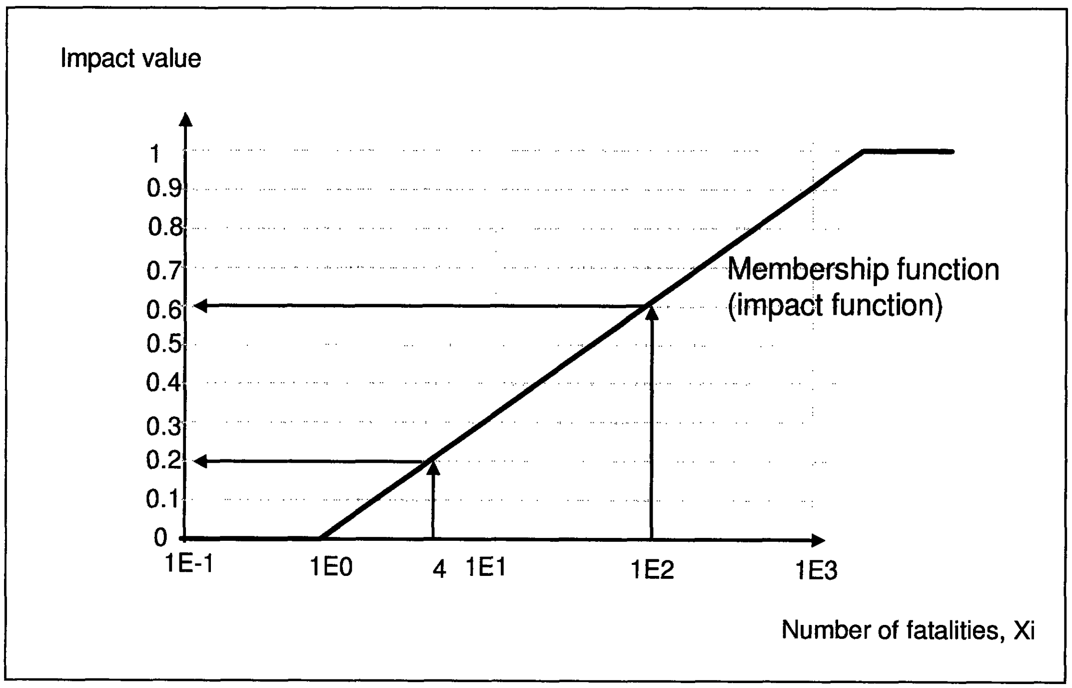

Figure 26: Membership function for total number of fatalities

(Bohnenblust et al, 1994)

The individual impact values must then be combined into a single impact value. The Swiss regulatory proposal does not give a definite answer on this issue, and just states that the maximum of all the indicators could be chosen as the overall impact value:

$$C = \max_i \{N_i\}.$$

Bohnenblust proposes a function value:

$$f_{p,A}(N_1, N_2, \ldots, N_9) = \min(1, \left((g_A^1(N_1))^p + \ldots + (g_A^9(N_9))^p\right)^{\frac{1}{p}},$$

where p is an integer parameter, derived from the Yager operator, and chosen equal to 5 by Bohnenblust in order to lead to a value more significant than the max value and less important than the sum of the individual impact values.

VI.C.2.b. Swiss criterion

Once a risk factor has been calculated, the acceptability of the risk must be addressed. The Swiss proposal includes a three-region approach: a region where risk is unacceptable, a region where it is acceptable and finally a region where risk must be reduced but in consideration of costs and benefits. Figure 27 presents the F-C curve:

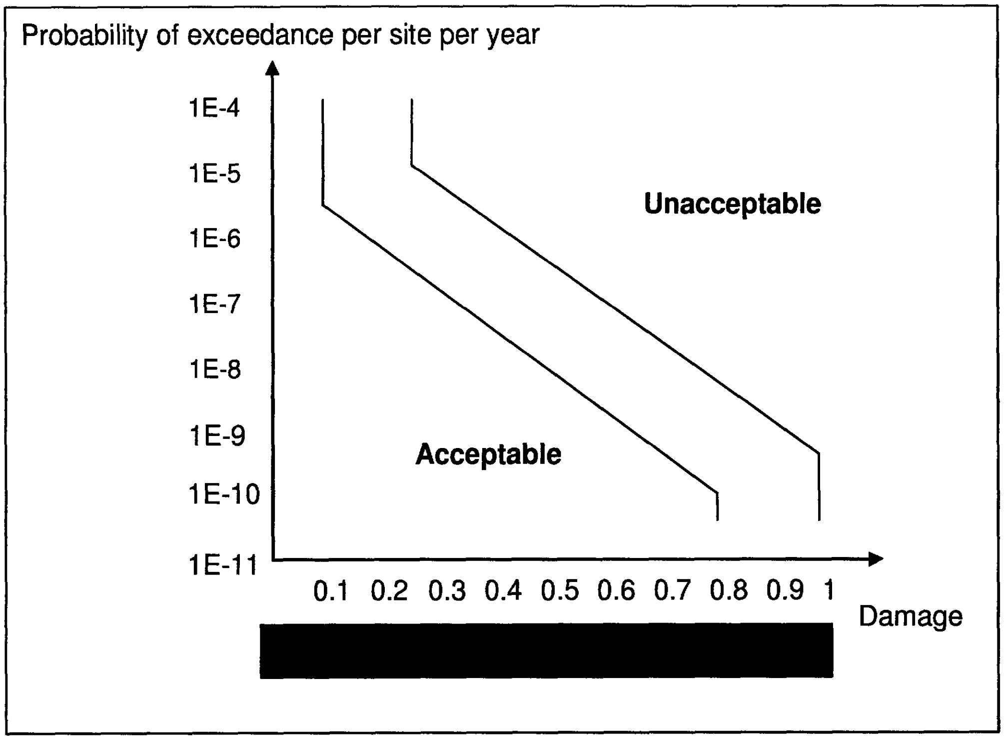

Figure 27: F-C curve proposed by the Swiss Ordinance

(Buwal, 1991)

A single value figure can also be used to compare previous accidents with very different consequences.

However, no consensus exists on the method in Switzerland and it is still discussed. One of the major issues of the method is that it involves subjective judgments almost at every step (definition of categories, translation of the consequences into individual impact values, combination of individual values into an overall value, definition of acceptable level of risk).

VI.D. Conclusion

Use of FN curves in the Netherlands has had a positive impact on safety. It is however hard to extrapolate these results to the United States since the two countries differ in geographical size, population density and in their number of reactors. Furthermore, societal risk from nuclear accidents should capture more than fatalities as a unique category of consequences. For instance, experience from the Chernobyl accident shows that consequences such as land contamination should also be included in any risk assessment tools aimed at limiting the societal risk from nuclear accidents. If "extended" measures of societal risk have been proposed, not one has ever been implemented. The Netherlands have announced their willingness to include such integrated measure in their environmental regulations, but no further details are currently available. The question of integrating such curve into the existing risk criteria in the U.S. has been recently asked.

Part VII. Should societal risk criteria be defined in the United States?

At least one member of the Advisory Committee on Reactor Safeguards (ACRS) in the United States has suggested establishing a F-C curve societal risk criterion for nuclear power plants, and has proposed to use as a consequence the overall societal consequences as determined by the total number of prompt fatalities, latent cancers, injuries, and land contamination (Kress, 2005).

The purpose of this part is to explore the question of societal risk criteria in the United States, to analyze the proposal, as well as to propose variations on the criterion.

VII.A. Description of proposal

VII.A.1. Overview

The F-C curves suggested by Kress are an extension of the "classical" FN curves (Kress, 2005). The ACRS member suggests using as a measure of consequence the overall societal cost as determined by the total number of prompt fatalities, latent cancers, injuries, and land contamination, all expressed in terms of dollars. For each of these four categories of consequences, level-3 PRAs are already able to produce complementary cumulative density functions (CCDF), with uncertainties accounted for, as illustrated in Figure 28.

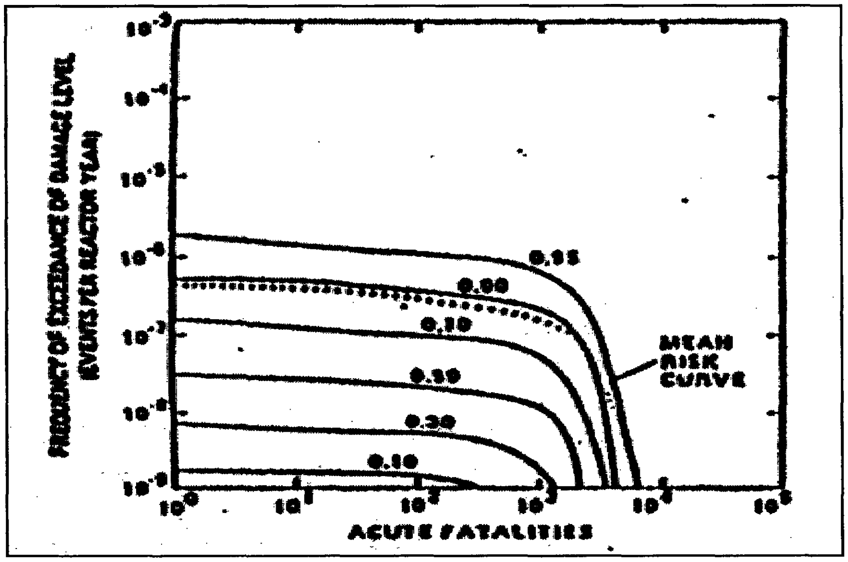

Figure 28: Complementary Cumulative Density Function for acute fatalities from Plant "X" level-3 PRA

CCDF for all four categories of consequences can be translated into a dollar value and then combined so as to obtain one curve capturing all consequences. Difficulties of such an approach are highlighted in Part VII.C.2. The tolerability of risk can then be assessed by comparing the curve to a criterion, such as the illustrative one proposed by Kress and reproduced in Figure 29.

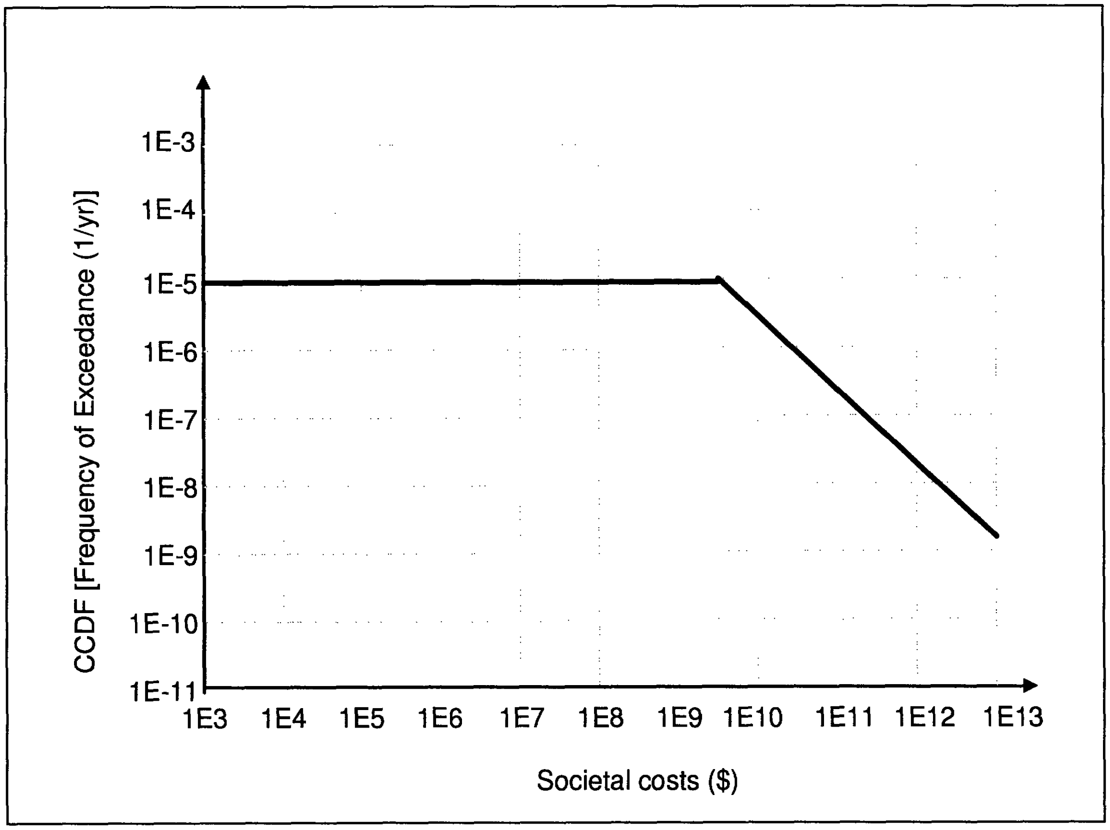

Figure 29: Illustrative Complementary Cumulative Density Function F-C risk acceptance criteria (Kress, 2005)

Kress's choice of the F-C curve shape is justified by the following arguments:

- Exactly like for FN curves, the area under the curve is equal to the expected cost in dollars per site and per year for all the categories of consequences previously defined. The expected cost, called "F-C cost-risk status", can be estimated for each plant, which allows the ranking of different plants based on that figure. If the estimated F-C curve is below the F-C criterion, then the area under the first is smaller than the area under the second. Hence, having a F-C curve criterion limits the F-C cost-risk status. Kress proposes as an example to set the F-C cost-risk status limit, i.e. the area under the curve, in a similar way used to define the QHOs: If there are 100,000 accidents per year in the U.S. and approximately 100 plants, and if the cost per death is taken equal to \$ 2.5 million, then the limit per plant should be set at 0.1% of the total cost of accidents, i.e.: $(0.001)*(2.5*10^6)*(1*10^5)/(100)=$ \$ $2.5*10^6$/site-yr. Since future plants are expected to be safer, the area under the F-C curve is equal to one tenth of this maximum cost-risk status, i.e. $ 2.5*10^5 /site-yr

- As the consequences tend to 0, the CCDF tends to the value of the Core Damage Frequency (or a preventive risk metric in the technology-neutral context). Therefore, the intersection of the F-C curve and the y-axis is an estimate of the CDF, and must be below the value of the intersection of the F-C criterion and the y-axis. Kress suggests using a value of the CDF limit equal to 10^{-5} /site-yr; a value coherent with the one that has been recently proposed by the USNRC (USNRC, 2006). Therefore, the asymptote of the curve at small consequences is equal to 10^{-5} /site-yr.

- Finally, Kress chooses to define the F-C curve in a risk-neutral manner

The additional criteria implied by the F-C curves are illustrated on Figure 30.

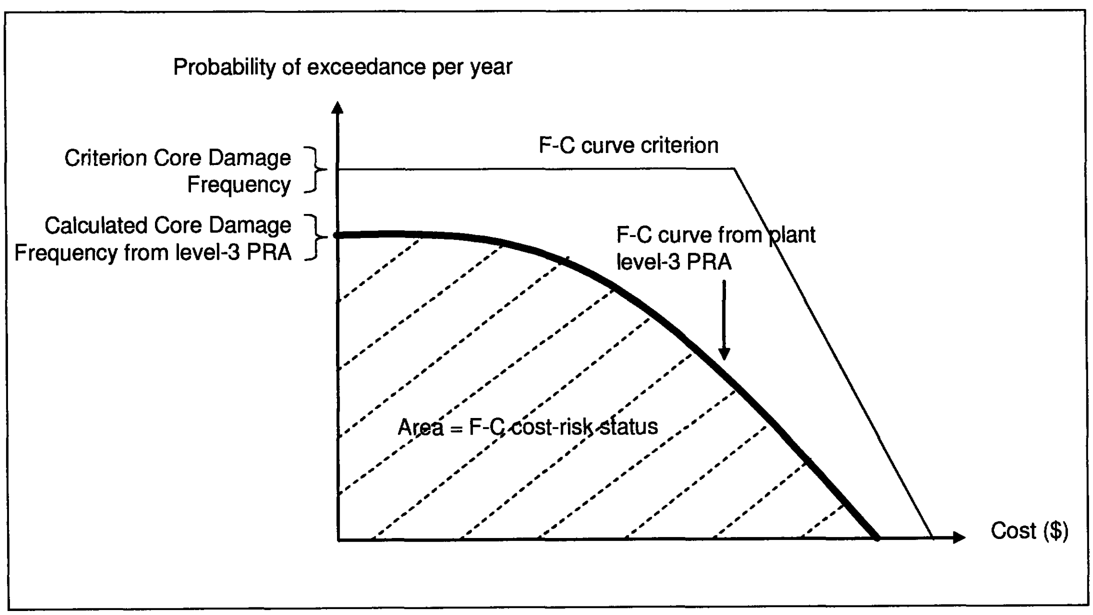

Figure 30: Criteria implied by the F-C curve

Furthermore, in order to account for uncertainties in the PRA calculations, a three-region approach has been proposed. Between the acceptable and the unacceptable regions, a cost-benefit improvement region can be defined. In that region, the costs and benefits of a change, all expressed in dollars, must be weighted in order to decide if the proposed change is acceptable. Each change is evaluated with regards to impact categories defined by the USNRC in the Regulatory Analysis Technical Evaluation Handbook (USNRC, 1997).

VII.A.2. Application of proposal

In order to calculate on a real case the societal risk from different power plants, Kress suggests using the results of the Generic Environmental Impact Statement for License Renewal of Nuclear Plants (USNRC, 1996), detailed in Appendix 1.

VII.A.2.a. Overview of the Generic Environmental Impact Statement for License Renewal of Nuclear Plants

The Generic Environmental Impact Statement (GEIS) examines wherever possible the environmental impacts that could occur as a result of renewing licenses of individual nuclear power plants. For that reason, it estimates the impact of postulated accidents and severe accidents on health effects, captured by early and latent fatalities, and off-site costs for the middle year of relicense (MYR) population for 74 power plants. The calculations are conservative, and no discount rate is considered here.

The GEIS assumes that the license renewal process will ensure that aging effects are controlled, i.e. that the probability of radioactive release from accidents will not increase over the license extension period. Most of the risk is assumed to be captured by the population around the plant, as well as the wind direction. This is a very restrictive assumption, which implies that societal risk due to plant characteristics is not accounted for (see Part V.D.2).
The Exposure Index (EI) methodology is used in NUREG-1437. The EI is a site-specific variable reflecting the population surrounding the plant, weighted by the site-specific wind direction frequency, which determines the fraction of population at risk

The CRAC computer code is used to calculate off-site severe accident costs for the area contaminated by the accident. The code estimates the evacuation costs, the value of crops contaminated and condemned, the value of milk contaminated and condemned, the costs of decontamination of property where practical, and the indirect costs resulting from the loss of use of property and income.

VII.A.2.b. Results

This method does not allow the construction of F-C curves. However, it provides estimates for the total number of early and latent fatalities as well as the off-site costs previously defined. To convert health effects into a monetary value, Kress uses 2.5 million dollars per fatality.

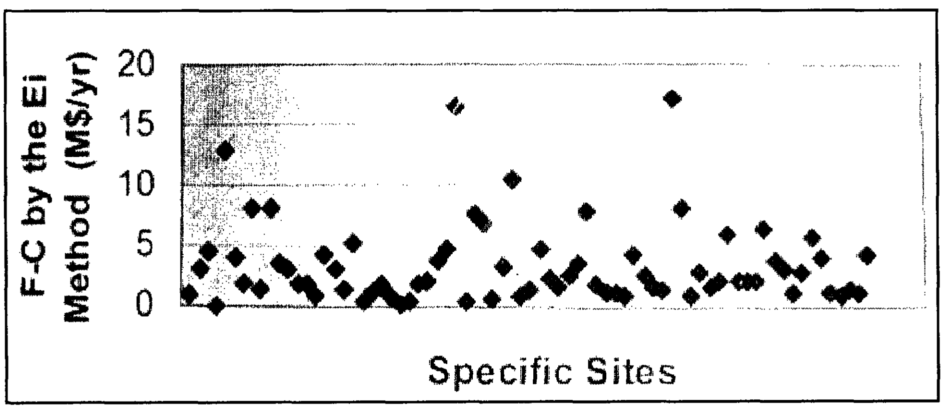

Figure 31: F-C risk-cost status (Kress, 2005)

The results are the following: two plants have a significant higher F-C risk-cost status than the others, with an F-C cost-risk status higher than 15 million dollar per reactor year. Two plants have an F-C cost-risk status between 10 and 15 million dollars per reactor year. The 70 remaining plants are below 10 million dollars.

As seen earlier, Kress suggests requiring that the total societal cost from nuclear accidents be less than 0.1 % of the total societal cost due to accidents in the U.S. , i.e. $ $2.5*10^5$ /site-yr for a value of life of 2.5 million dollars, and considering advanced plants will be ten times safer than current plants.

There are many outliers with this criterion. However, even if no criterion is used, it is still possible to observe a wide range of F-C cost-risk status and that the value is significantly higher than the others for a few of them.

This risk measure relies on the value of statistical life chosen. It is thus necessary to evaluate the dependence of the results on such value.

VII.B. Valuation of life is required in this approach

VII.B.1. USNRC policy regarding valuation of life

Defining the overall societal risk as the sum of the fatalities, injuries and land contamination implies valuing explicitly human life, which is a controversial issue. The position of the USNRC on that matter is stated in NUREG/BR-0184, which is the regulatory analysis technical evaluation handbook (USNRC, 1997). For cost-benefit analyses, the USNRC recommends using the monetary equivalent of $2000/person-rem for accidental and routine emissions, for both public and occupational exposure, and taking into account all the accident-related health effects.

The mean cancer risk factor reported in the literature is 5×10^{-4}/rem, and the range of uncertainties is estimated to be $3\times10^{-4}-9\times10^{-4}$ (Guenther and Thein, 1997). This cancer risk factor value accounts for the fact that the young have an increased sensitivity to radiations, the non-fatal cancers and the severe genetic effects.

The statistical value of life for latent fatalities entailed by the USNRC guideline is therefore 4,000,000 dollars and the range of uncertainties is $ 2,000,000 - $ 7,000,000. Based on this uncertainty, the range of values used to assess the strength of the results is chosen equal to $ 1,000,000 - $ 10,000,000.

Literature is scarce on how a latent fatality should be weighted in comparison to an early fatality. A value for early fatality five times higher than the value for latent fatality has been used in a societal risk proposal (Okrent, 1981). However, no rationale for such figure is provided. We will assume in the following calculations that the statistical value of life for an early fatality is at least as high as that of a latent fatality.

One should note that new methods are being developed to replace the traditional concept of a calculation based on the Value of Statistical Life with an evaluation of the Value of Life Year Lost. This concept would be particularly useful to weight an early death against a latent death.

VII.B.2. Sensitivity analysis

Using NUREG-1437 data, it is possible to assess the importance of latent fatalities with regards to early fatalities and to estimate the F-C cost-risk status for the 74 plants using different values of statistical life. Each plant is defined by a number between 1 and 74.

VII.B.2.a. Latent fatalities dominate

The ratio of the predicted number of early fatalities by the predicted number of latent fatalities can be calculated for each plant using NUREG-1437 data, as shown in Figure 32.

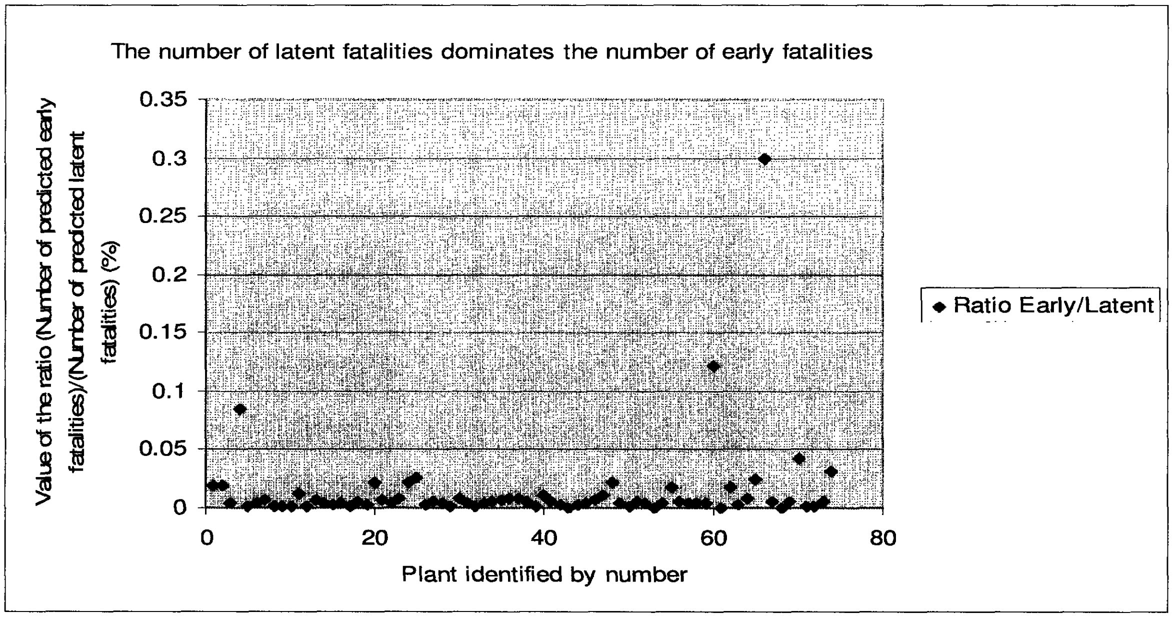

Figure 32: Ratio of the predicted number of early fatalities to the predicted number of latent fatalities

For 71 plants, this ratio is below 5%; for 2 plants, the ratio lies between 5 and 15%. Finally, for only one plant is the ratio as high as 30%. Therefore, latent fatalities dominate early fatalities in terms of absolute predicted numbers.

VII.B.2.b. Frequency-consequence cost-risk status

If we assume that a statistical value of life can be calculated (methods are presented in Appendix 2), the strength of F-C risk measure can be assessed by analyzing the dependence of the results on the value of life chosen. Values ranging from $1,000,000 $ to $10,000,000 are chosen.

- Case 1: Value for early and latent fatality is $ 1,000,000

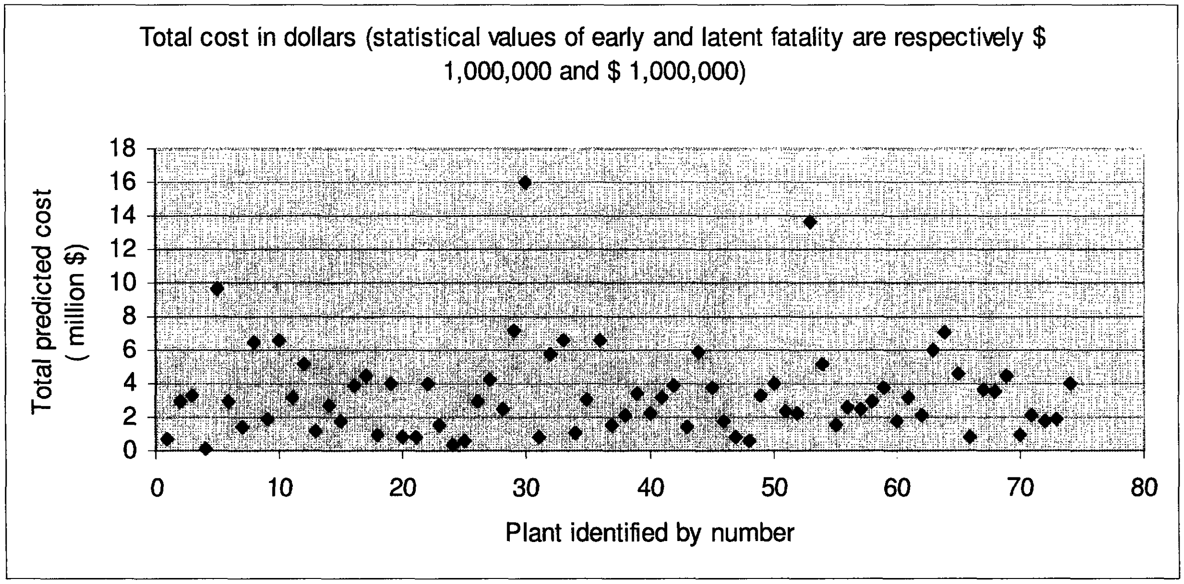

Figure 33: Frequency-consequence cost-risk status - value for early and latent fatality is $ 1,000,000

The three plants with the highest F-C cost-risk status are plants 30, 53 and 5, from the highest to the lowest.

- Case 2: Value for early fatality is $2,500,000 and value for latent fatality is $1,000,000

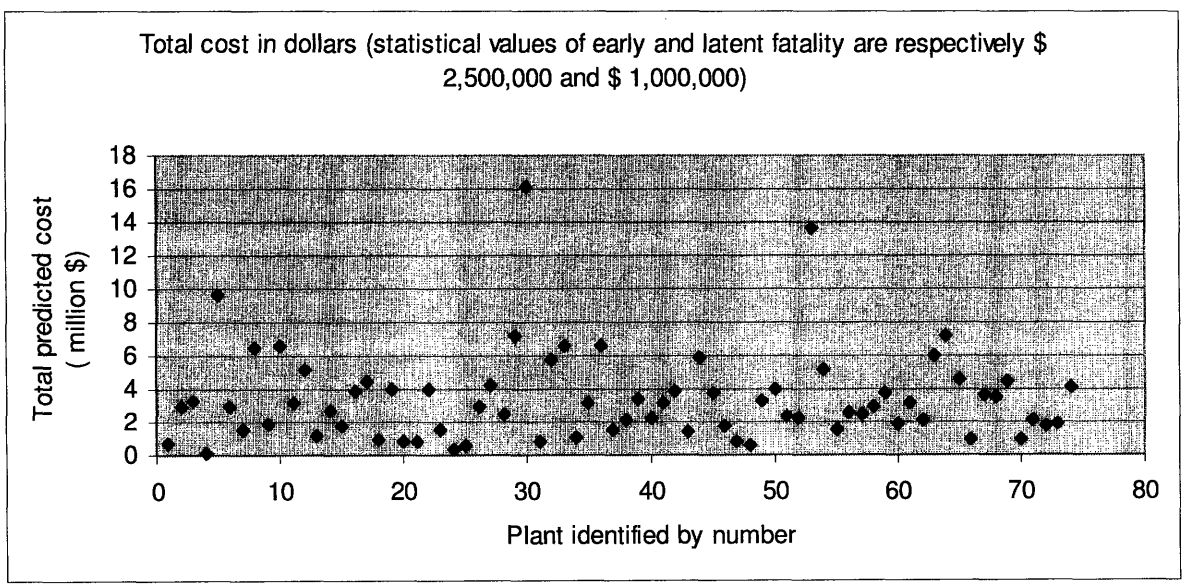

Figure 34: Frequency-consequence cost-risk status – Value for early fatality is $2,500,000 and value for latent fatality is $1,000,000

The three plants with the highest F-C cost-risk status are plants 30, 53 and 5, from the highest to the lowest.

- Case 3: Value for early fatality is $2,500,000 and value for latent fatality is $2,500,000

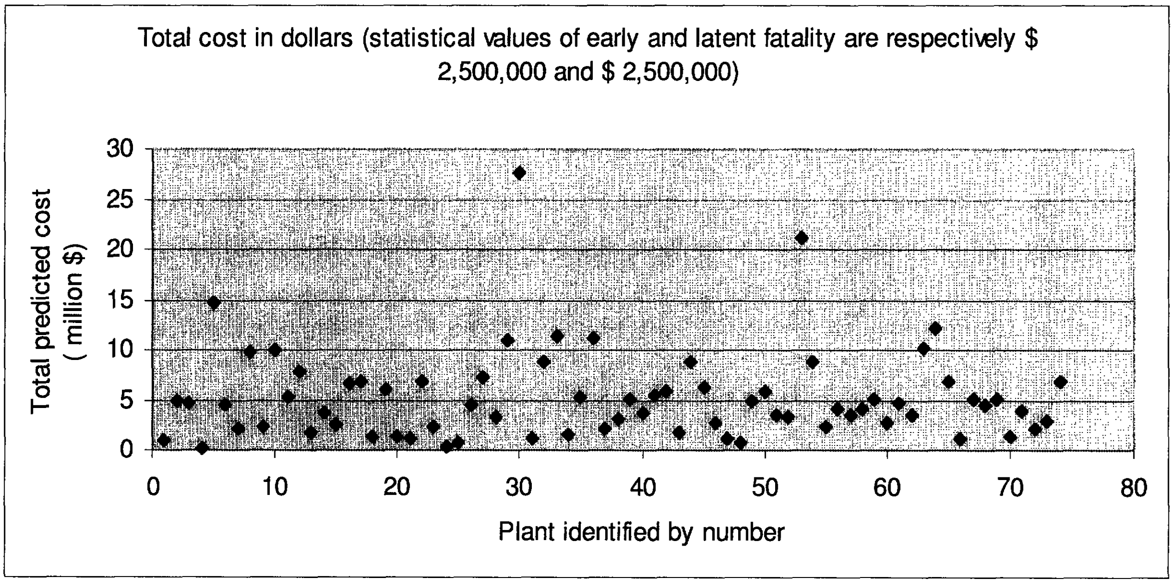

Figure 35: Frequency-consequence cost-risk status - Value for early fatality is $2,500,000 and value for latent fatality is $2,500,000

The three plants with the highest F-C cost-risk status are plants 30, 53 and 5, from the highest to the lowest.

- Case 4: Value for early fatality is $4,000,000 and value for latent fatality is $4,000,000

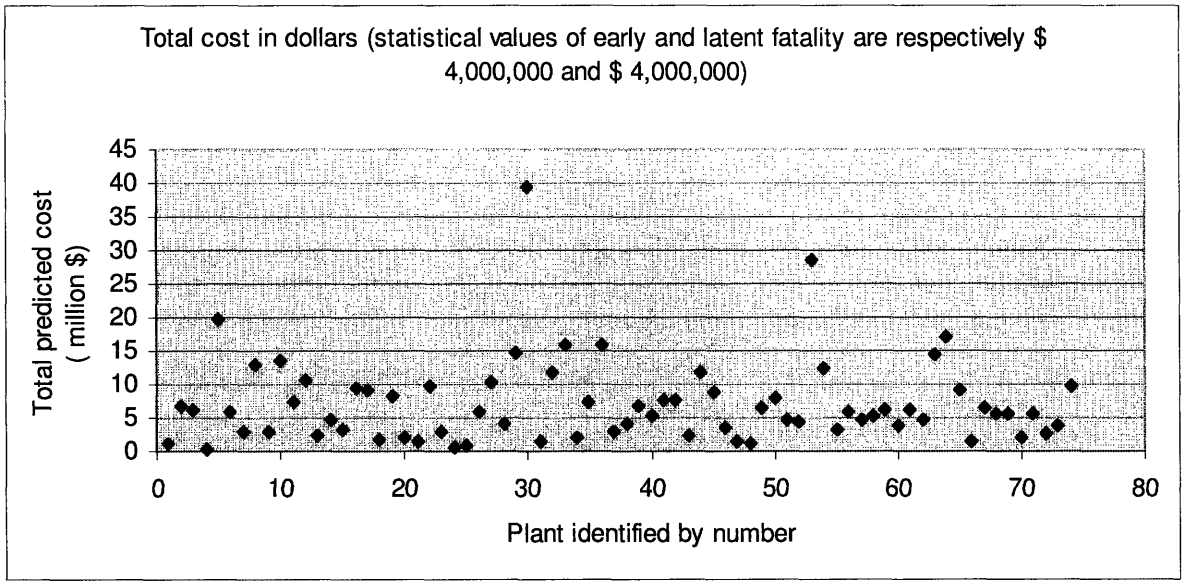

Figure 36: Frequency-consequence cost-risk status – Value for early fatality is $4,000,000 and value for latent fatality is $4,000,000

The three plants with the highest F-C cost-risk status are plants 30, 53 and 5, from the highest to the lowest.

- Case 5: Value for early fatality is $12,500,000 and value for latent fatality is $2.5,000,000

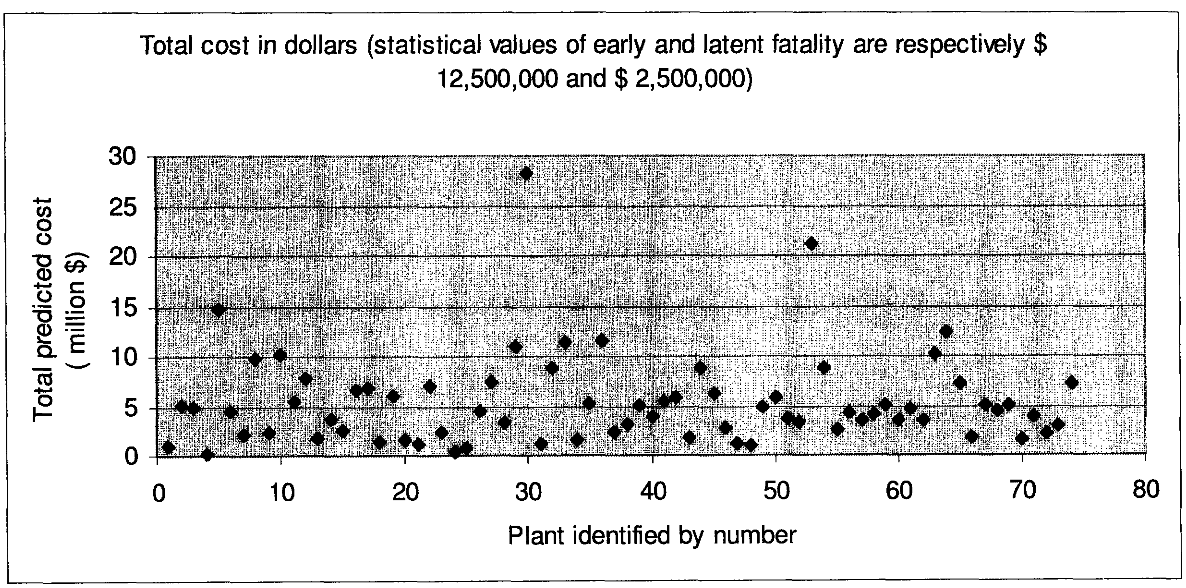

Figure 37: Frequency-consequence cost-risk status - Value for early fatality is $12,500,000 and value for latent fatality is $2,500,000

The three plants with the highest F-C cost-risk status are plants 30, 53 and 5, from the highest to the lowest.

Case 6: Value for early fatality is $20,000,000 and value for latent fatality is $4,000,000

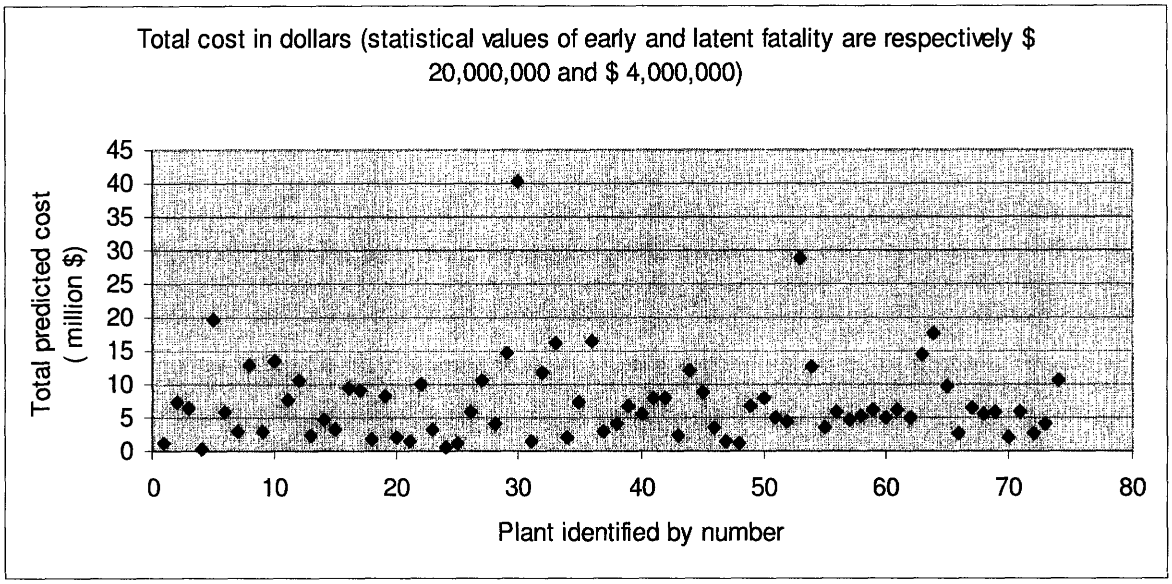

Figure 38: Frequency-consequence cost-risk status - Value for early fatality is $20,000,000 and value for latent fatality is $4,000,000

The three plants with the highest F-C cost-risk status are plants 30, 53 and 5, from the highest to the lowest.

VII.B.3. Summary of results and implications

The previous results are summarized in Table 6:

Case	Value of life (early fatality) In million dollars	Value of life (latent fatality) In million dollars	Plants with highest F-C risk-cost status
1	1	1	30, 53,5 (in order)
2	2.5	1	30, 53,5 (in order)

3	2.5	2.5	30, 53,5 (in order)
4	4	4	30, 53,5 (in order)
5	12.5	2.5	30, 53,5 (in order)
6	20	4	30, 53,5 (in order)

Table 6: Ranking of plants based on their overall societal cost for different values of statistical life

- In light of the results, it appears that no matter the statistical values of life chosen for early and latent fatalities, there are always 3 plants whose F-C cost-risk status is significantly higher than those of the remaining 71 plants. Since the effects of plant aging were not accounted for in the calculations, we can conclude that most of the risk comes from an increased number of people living around the plant, as well as an increase of the off-site costs of accidents (for instance increase in the price of land, crop values, or real estate). The ratio between the costs due to fatalities, both early and latent, and off-site costs, depends of course on the value of life chosen. The following figure presents the calculation of the ratio (Off-site costs in dollars)/(Predicted early and fatality costs for statistical values of early and latent fatalities respectively equal to $12,500,000 and $2,500,000).

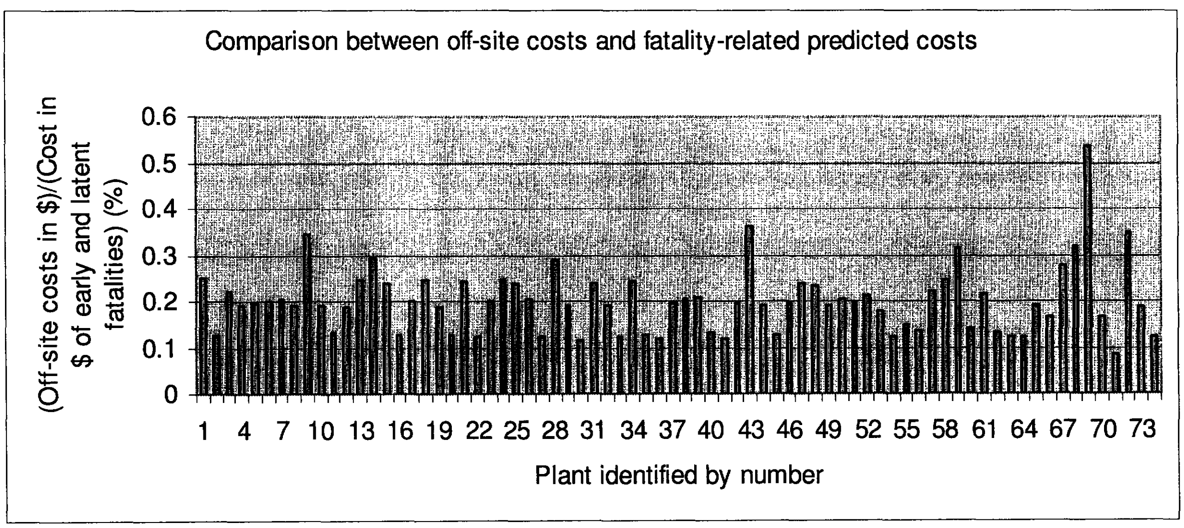

Figure 39: Comparison between off-site costs and fatality-related predicted costs (statistical values of early and latent fatalities respectively equal to $12,500,000 and $2,500,000)

For the majority of cases, the off-site costs represent around 20% of the fatality-related costs. This ratio decreases if higher statistical values of life are chosen. Therefore, costs due to latent fatalities dominate the overall predicted societal cost.

- The criterion proposed by Kress (when not divided by 10 to account for the fact that the plants considered in NUREG-1437 are current plants, and not advanced plants) shows a large number of outliers. Following up on Kress's idea, we can build a criterion similarly to what has been done with the QHOs: There are approximately 100,000 accidental deaths in the U.S. per year, and most of these deaths are early fatalities. The criterion should therefore be calculated using the statistical value of life for early fatality.

In the case where we valued an early life to be equal to 12.5 million dollars and a latent fatality to 2.5 million dollars, the criterion becomes:

$(0.001)*(12.5*10^6)*(1*10^5)/(100)=\$\ 12.5*10^6$/site-yr.

As illustrated on Figure 40, there are only three unambiguous outliers using this criterion:

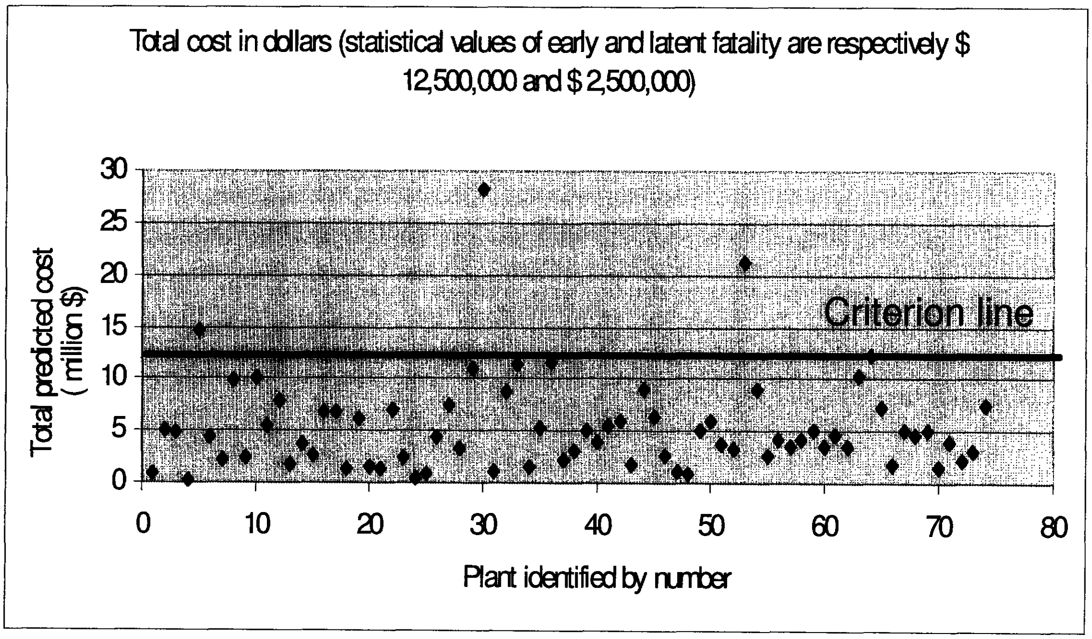

Figure 40: Assessment of the tolerability of F-C cost risk status using statistical values of early and latent fatalities respectively equal to \$12,500,000 and \$2,500,000

The same can be done for a statistical value of early fatality equal to \$20,000,000 and a value of latent fatality equal to 4,000,000.

The criterion becomes: $(0.001)*(20*10^6)*(1*10^5)/(100)=\$\ 20*10^6$/site-yr, and the tolerability of societal risk in that case is illustrated on Figure 41.

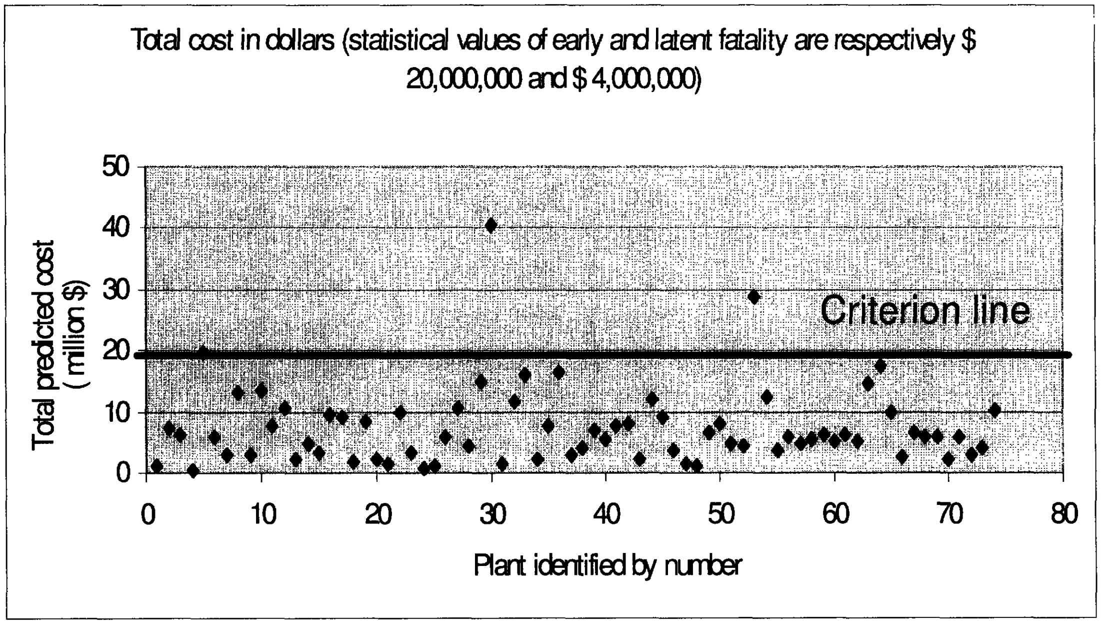

Figure 41: Assessment of the tolerability of F-C cost risk status using statistical values of early and latent fatalities respectively equal to $20,000,000 and $4,000,000

- The previous calculations show that societal risk is unequally distributed in the U.S., since some plants involve a much higher societal cost than others. This could be used by regulatory authorities as a screening criterion: outlying plants should be scrutinized in special cases, for instance when licensees require power uprates.

VII.C. Issued related to the approach

VII.C.1. Valuation of injuries

Kress proposes to define the societal cost of nuclear accidents as the sum of the costs of early fatalities, latent fatalities, land contamination and injuries. If there is available literature on the valuation of life, be it to support it or to criticize it (Heinzerling et al, 2002), data on valuation of injuries is very scarce; which makes it difficult to include injuries in the measure.

VII.C.2. Correlation of variables

The consequences "early fatalities", "latent fatalities", "land contamination" and "injuries" can be treated as random variables. For each of them, it is possible to obtain a complementary cumulative density function as an output of a level-3 PRA. To obtain a monetary equivalent for early and latent fatalities, one can easily multiply the consequence axis by the statistical value of life. However, building an aggregated risk curve for all the consequences requires the knowledge of the correlation between the different random variables. For instance, the more people are evacuated (cost taken into account in the off-site cost category or land contamination), the lower are the health effects (the cost of early/latent fatalities decreases). Estimate of these correlations requires additional burdensome and uncertain calculations. This hasn't been done up to date.

VII.C.3. Maturity of computer codes

Computer codes such as Melcor Accident Consequence Code System (MACCS) used for level 3-PRAs produce risk curves for, among other consequences, early fatalities, cancers, injuries, collective dose, and offsite property damage. The epistemic uncertainties are very high, especially for the first three items, as illustrated on figure 42:

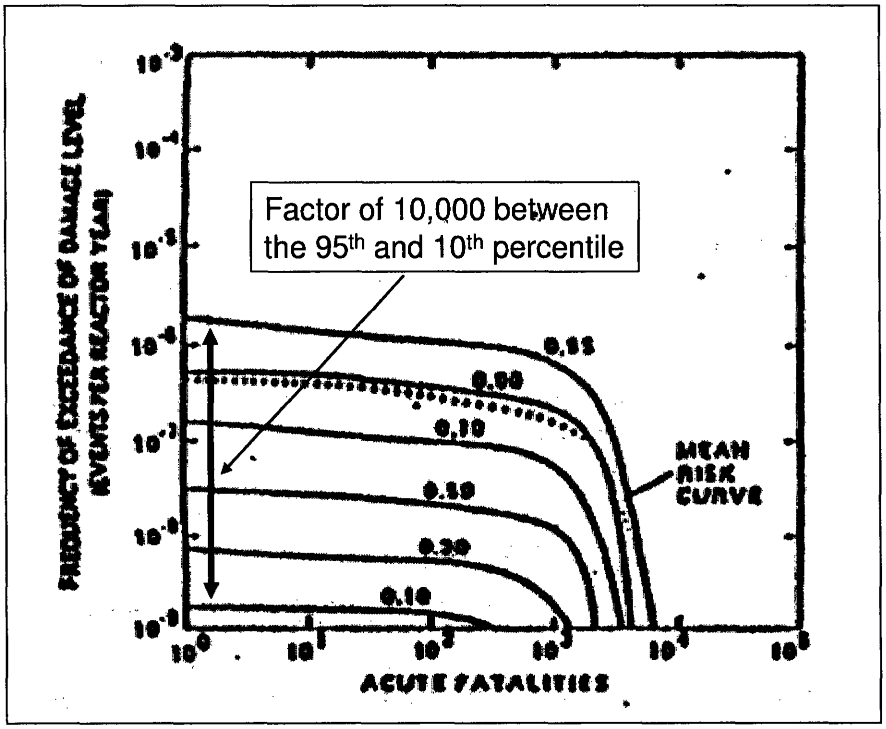

Figure 42: High uncertainties for Plant "X" level-3 PRA output

Decision-making with such uncertainties is highly complex, and tools must be improved. A group of experts of the Nuclear Energy Agency (NEA) recommended in 2000 that accident consequence assessment codes be further developed (NEA, 2000).

VII.C.4. Siting vs Design

Implementing a societal risk criterion is complex because it requires both the knowledge of precise details on the site where the plant is located (e.g. wind direction, density and location of population, evacuation resources) and the plant characteristics. For that reason, it is very unlikely that such criterion could be part of the licensing process, since the designer has little knowledge of the site where the plant will be located.

The following figure details what data is necessary depending on the definition of the F-C curve. The consequence chosen by Farmer in 1967 was the amount of Iodine 131 released. This consequence measure did not require knowledge of the site. The F-C curve developed by the USNRC for the selection of Licensing Basis Events uses the dose to an individual at a specific distance from the site as a consequence measure. Only weather data or models are needed to calculate this consequence. This is not the case for the F-C societal risk criterion which requires both the knowledge of the site and the plant.

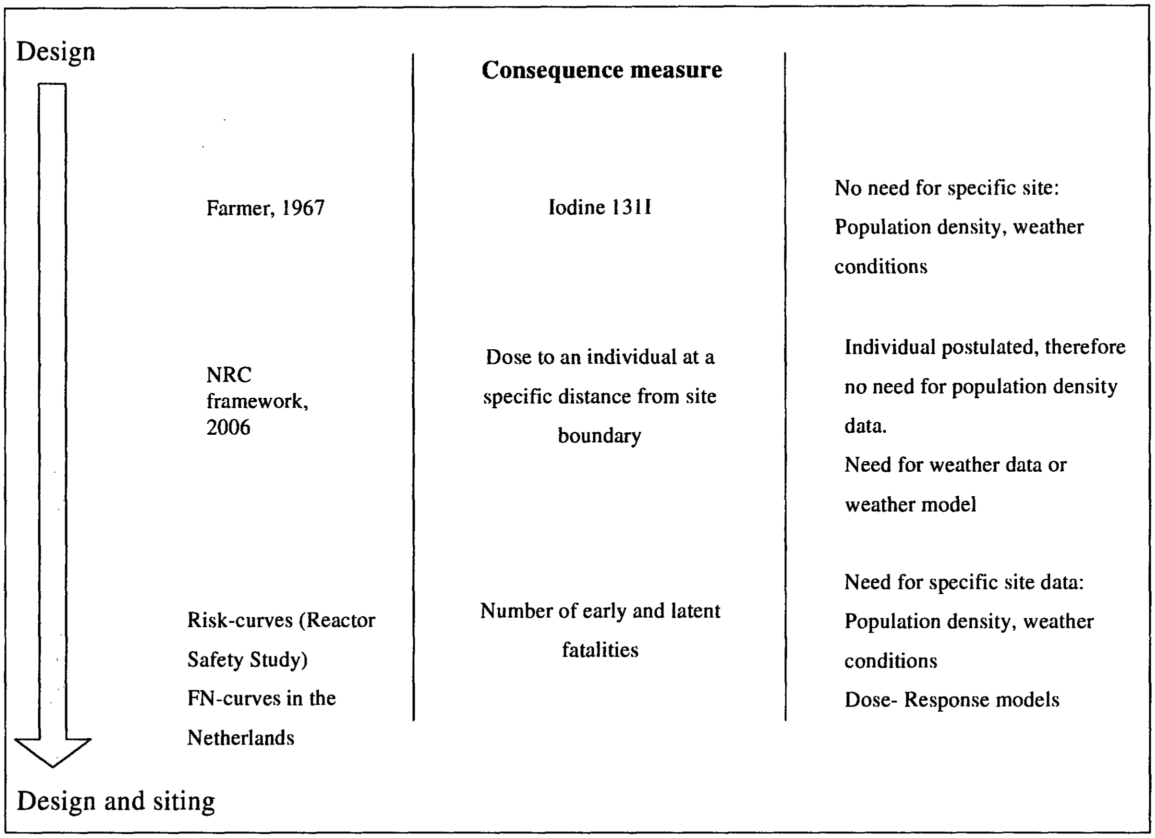

Figure 43: Required data depending on the type of frequency-consequence curve

VII.D. Conclusion

A societal risk criterion defined in the form of a F-C curve would be a useful way to control the risk of current reactors. Indeed, the fleet of reactors displays a wide range of F-C cost-risk statuses and certain plants should be closely scrutinized, for instance when they request power uprates. However, changing the regulations now for current plants would be acknowledging that the point has been missed for decades. Kress suggests using such curves for advanced reactors. Several objections can be made to this suggestion: first, the use of such criterion requires the knowledge of both the site (for instance population and weather) and plant characteristics at the time of licensing; which is often not the case. Second, advanced plants are expected to be so safe that such criterion may not be needed. A new evaluation of the need for such criterion should be done when data on level-3 PRAs of Generation IV reactors becomes available.

Part VIII. Summary of conclusions

The licensing of nuclear power plants has focused until now on Light Water Reactors and has not incorporated systematically insights and benefits from PRA. With the goal of making the licensing process more efficient, predictable and stable for advanced reactors, the U.S. Nuclear Regulatory Commission has recently drafted a risk-informed and technology-neutral framework for new plant licensing. The Commission expects that advanced nuclear power plants will show enhanced margins of safety, and that advanced reactor designs will comply with the Commission's Safety Goal Policy Statement. In order to meet these expectations, PRA tools are currently being considered; among them are frequency-consequence curves, which plot the frequency of having C or more consequences (fatalities, injuries, dollars, dose...) against the consequences C. The objective of this thesis is to study their role and usefulness in the context of the new NRC framework, as well as to explore their potential application as a societal risk acceptance criterion.

In parts II, III and IV, we have presented and analyzed F-C curves, as defined by the USNRC, and concluded that such risk assessment tools contributed effectively to the definition of a risk-informed licensing process, for they allowed, among other changes, the implementation of structuralist and rationalist Defense-in-Depth. Furthermore, the use of F-C curves introduces a major change in the regulations by defining a systematic selection process of Licensing Basis Events, intended to replace the fully deterministic Design-Basis Accidents.

The USNRC's use of F-C curves is based on individual risk and is therefore quite innovative, since these tools are classically used to assess societal risk.

In part IV and V, we introduced the general concept of societal risk and the quantitative tools available to assess and limit such risk. Our conclusions can be summarized as follows:

- Existing tools concentrate on a single type of consequence, in general early fatalities, and are known as FN curves. Those curves have entered the regulations in the Netherlands, and have had a positive impact on safety. It is however hard to extrapolate these results to the United States since the two countries differ in geographical size, population density and in their number of reactors.

- An overview of nuclear accidents shows that societal risk from nuclear accidents should capture more than fatalities as a unique category of consequences. Among other categories, any risk measure specific to the nuclear field should include latent fatalities and land contamination. If "extended" measures of societal risk have been proposed, not one has ever been implemented. The question of integrating such curve into the existing risk criteria in the U.S. has been recently raised.

Finally, after a review in Part VII of the latest proposal to include an extended societal risk criterion in the U.S., we concluded that:

- Societal risk is affected by the siting of the nuclear power plant and the amount of radioactive material present in the core, and not by the design of the reactor. Changing design to suit the site defeats the purpose of standardization and the public would want all sites to have the best available design.
- Current plants involve a wide range of societal costs, and certain costs were deemed unacceptable when compared to the criteria we defined.
- Plants that are considered as outliers in our model should be closely scrutinized when requesting power uprates, likely to increase the amount of radioactive material in the core.
- In light of the available data, societal risk criteria are not needed for future plants. A new evaluation of the need for such criteria should be done when data on level-3 PRAs of Generation IV reactors becomes available.

List of references

Ale, B.J.M., Laheij, G.M.H., Uijt De Haag, P.A.M., 1996, Zoning instruments for major accident prevention, Probabilistic Safety Assessment and Management, ESREL 96 – PSAM III, Crete, 1996, p.1911

Ale, B.J.M, 2005, Tolerable or acceptable: A comparison of risk regulation in the United Kingdom and in the Netherlands, Risk Analysis, Vol. 25, No. 2

Apostolakis, G.E., 1990, The concept of probability in safety assessments of technological systems, Science, New Series, Vol. 250, No. 4986 , pp. 1359-1364

Apostolakis, G.E., 1993, Proceedings of Workshop on Model Uncertainty: Its Characterization and Quantification, A. Mosleh, N. Siu, C. Smidts, and C. Lui, Eds., Annapolis, MD, October 20-22, 1993, Center for Reliability Engineering, University of Maryland, College Park, MD

Apostolakis G.E., 2000, Interview, "Apostolakis: On PRA", Nuclear News, March 2000

Ball D. J. and Floyd P. J., 1998, Societal Risk, Crown Copyright, Health & Safety Executive, Risk Assessment Policy Unit, London

Bohnenblust, H., Christen, P., Seitz, S., 1994, A methodology for assessing catastrophic damage to the population and environment: A quantitative multi-attribute approach for risk analysis based on fuzzy set theory, Process Safety Progress, Vol. 13, No. 4 pp. 234-238

BUWAL (Bundesamt für Umwelt, Wald und Landschaft, Swiss Agency for the Environment, Forests and Landscape), 1991, Manuel I «Ordonnance sur les accidents majeurs OPAM », Switzerland

Cazzoli, E., Khatib-Rahbar M., Schmocker, U., Isaak H.P., 1993, Approach to quantification of uncertainties in Probabilistic Safety Assessment, Proceedings of the European Safety and Reliability Conference, 10-12 May 1993, Munich, Germany

Chanin, D. et al., 1990, Melcor Accident Consequence Code System (MACCS): User's Guide, NUREG/CR-4691, Sandia National Laboratories

Evans, A.E.,Verlander, N.Q., 1997, What is wrong with criterion FN lines for judging the tolerability of risk?, Risk Analysis, Vol. 17, No. 2, pp. 157-168

Evans, 2003, Transport Fatal Accidents and FN curves: 1967-2001, HSE Research Report 073, HSE Books, Sudbury, Suffolk

Farmer, F.R., 1967, Siting criteria – A New Approach, Proceedings of IAEA symposium on the containment and siting of nuclear power reactors, Vienna, 3–7 April 1967, Paper IAEA-SM89/34, pp. 303-329

Freeman, M., 2001, Who Killed U.S. Nuclear Power?, 21st Century Science & Technology, Spring 2001 , http://www.21stcenturysciencetech.com/articles/spring01/nuclear_power.html, last consulted 12/07/06

Guenther, C. F., Thein, C., 1997, Estimated Cost of Person-Sv Exposure, Health Physics, Vol. 72, No. 2 pp. 204-221

Health and Safety Executive, 1992, The Tolerability of Risk from Nuclear Power Stations, Revision 1992, available on the HSE website: http://www.hse.gov.uk/nuclear/tolerability.pdf, last consulted 12/07/06

Heinzerling, L., Ackerman, F., 2002, Pricing the priceless: Cost-Benefit Analysis of Environmental Protection, Georgetown Environmental Law and Policy Institute, Georgetown University Law Center

Hirschberg, S., Strupczewski, A., 1999, Comparison of accident risks in different energy systems, How acceptable?, International Atomic Energy Agency Bulletin

Houts, P. S., Cleary P. D., Hu T., 1988, The Three Mile Island crisis :psychological, social, and economic impacts on the surrounding population, University Park : Pennsylvania State University Press

Institute of Chemical Engineering, 1985, Nomenclature for hazard and risk assessment in the process industries

Jonkman, S.N., Van Gelder P.H.A.J.M., Vrijling J.K., 2003, An Overview of Quantative Risk Measures, Journal of Hazardous Materials, Vol. 99, No.1 pp. 1-30

Keeney R.L., quoted by Slovic, 1980, Evaluating Alternatives Involving Multiple Fatalities, Operations Research, Vol. 28, pp. 188-205

Keeney, R.L. , Von Winterfeldt, 1994, D., Managing nuclear waste from power plants, Risk Analysis, Vol. 14, No. 1 pp. 107-130

Komanoff, C., 1981, Power Plant Cost Escalation, Published by Komanoff Energy Associates

Kress, T.S., 2005, Frequency-consequence Risk Acceptance Criteria, International Topical Meeting on Probabilistic Safety Analysis PSA '05, September 11-15 2005, San Francisco, California

Nuclear Energy Agency, 2000, Methodologies for Assessing the Economic Consequences of Nuclear Reactor Accidents

Okrent, D., 1981, Nuclear reactor safety: on the history of the regulatory process, Madison, Wisconsin: University of Wisconsin Press

Okrent, D., Dunster, H. J., Van Reijen G., 1981, Industrial Risks [and Discussion], Proceedings of the Royal Society of London, Series A, Mathematical and Physical Sciences, Vol. 376, No. 1764, The Assessment and Perception of Risk (Apr. 30, 1981), pp. 133-149

Saji G., 2003, A new approach to reactor safety goals in the framework of INES, Reliability Engineering & System Safety, Vol. 80, No. 2 pp. 143-161

Slovic P., Lichtenstein S., Fischhoff B., 1984, Modeling the Societal Impact of Fatal Accidents, Management Science, Vol. 30, No.4

Sorensen J. N., Apostolakis G. E., Kress T. S., Powers D. A., 1999, On the Role of Defense in Depth in Risk- Informed Regulation" , presented at PSA 1999, August 22-25, 1999

USNRC, 1975, Reactor Safety Study: An Assessment of Accident Risks In U.S. Commercial Nuclear Power Plants, WASH-1400 (NUREG-75/014)

USNRC, 1986, Safety Goals for the Operation of Nuclear Power Plants; Policy Statement, Federal Register, Vol. 51, p. 30028, August 21, 1986

USNRC, 1994, Commission Policy Statement on the Regulation of Advanced Nuclear Power Plants, 59 FR 35461

USNRC, 1995, Use of probabilistic risk assessment methods in nuclear activities: Final policy statement, 60 FR 42622, Federal Register, Vol. 60

USNRC, 1996, Generic Environmental Impact Statement for License Renewal of Nuclear Plants, NUREG-1437, Vol.1

USNRC, 1997, NUREG/BR-0184: Regulatory Analysis Technical Evaluation Handbook

USNRC, 1999, Commission's White Paper on "Risk-Informed and Performance-Based Regulation"

USNRC, 2004, Regulatory Structure for New Plant Licensing, Part 1: Technology-Neutral Framework, Working Draft Report, Revision a, December 2004

USNRC, 2004, SECY-04-0037, Issues related to proposed rulemaking to risk-informed requirements related to large break loss-of-coolant accident (OCA) break size and plans for rulemaking on LOCA with coincident loss-of-offsite power

USNRC, 2004, Regulatory Analysis Guidelines of the US Nuclear Regulatory Commission (NUREG/BR-0058), Rev. 4

USNRC, 2006, Nuclear Regulatory Commission website, Glossary http://www.nrc.gov/reading-rm/basic-ref/glossary.html, last consulted 12/07/06

USNRC, 2006, Fact Sheet on the Accident at Three Mile Island http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/3mile-isle.html, last consulted 12/07/06

USNRC, 2006, Working draft, Framework for development of a risk-informed performance-based, technology-neutral alternative to 10 CFR Part 50

Versteeg, M.F., 1992, Showing compliance with probabilistic safety criteria and objectives, Reliability Engineering and System Safety, Vol. 35 pp. 39-48

Viscusi W.K., Vernon, J.M., Harrington J.E., 2000, Economics of Regulations and Antitrust, Third Edition, The MIT Press Cambridge, Massachusetts

Vrijling et al, 1995, A framework for risk evaluation, Journal of Hazardous Material, Vol. 43 pp. 245-261

Vrijling J.K., Van Gelder P.H.A.J.M, 1997, Societal Risk and the Concept of Risk Aversion, In: C. Guedes Soares (Ed.), Advances In Safety and Reliability, Vol.1, Lissabon, pp.45-52

Vrijling J.K. et al, 2006, FN criteria for risk-regulation and probabilistic design, Proceedings of the 8th International Conference on Probabilistic Safety Assessment and Management, May 14-18, 2006, New Orleans, Louisiana, USA

VROM, The Netherlands Ministry of Housing, Spatial Planning and the Environment, 2005, Convention on Nuclear Safety, Dutch National Report

VROM, The Netherlands Ministry of Housing, Spatial Planning and the Environment, 2006, Information Sheet on External Safety Policy, Societal Risk, http://international.vrom.nl/docs/internationaal/societal%20risk.pdf, last consulted on 06/09/2006

Winkler, R.L., 1996, Uncertainty in probabilistic risk assessment, Reliability Engineering and System Safety, Vol. 54, No. 2 pp. 127-132

Appendix 1: Overview of NUREG-1437

Background

Operating licenses of nuclear power plants may be renewed for up to 20 years beyond the 40-year term of the initial license. Such renewal is authorized by the Atomic Energy Act of 1954 and the renewal process examines if the plant can continue to operate safety during the extension period. Limiting the initial operating license to 40 years was justified for economic and antitrust considerations. The first operating license will expire in 2009, and 40 % of the operating licenses will expire by 2015. In 1991, the USNRC published safety requirements for license renewal as 10 CFR Part 54. This first license renewal rule was amended in 1995. The operator that wishes to renew its license must submit a report that identifies the systems, structures and components that would be affected by the license renewal, shows that the effect of aging are well managed; and finally analyzes the environmental impact of the renewal (the scope of the environmental review is codified in 10 CFR 51). Independent reviews by the USNRC and the ACRS are carried out.

The Generic Environmental Impact Statement

The Generic Environmental Impact Statement (GEIS) examines wherever possible the environmental impacts that could occur as a result of renewing licenses of individual nuclear power plants. The GEIS was undertaken to provide the technical basis for an amendment to the 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," with regard to the renewal of nuclear power plant operating licenses.

Assumptions

On a high-level, the increase in risk during the renewal period can be due to either deterioration of the plant safety itself due to aging phenomena for instance; or in the change in the environment around the plant (e.g. increase in the density around the plant). The GEIS assumes that the license renewal process will ensure that aging effects are controlled, i.e. that the probability of radioactive release from accidents will not increase over the license extension period. Most of the risk is assumed to be captured by the population around the plant, as well as by the wind direction.

Methodology for predicting risk

Both the risks from design-basis accidents and severe accidents are evaluated in the GEIS. Doses and the resulting health effects, captured by early and latent fatalities, are estimated for the middle year of relicense (MYR) population, defined as the "estimated midpoint of the renewal period for a given plant rounded upward to the next year of available population data".
The Exposure Index (EI) methodology was used. The EI is a site-specific variable reflecting the population surrounding the plant, weighted by the site-specific wind direction frequency, which

determines the fraction of population at risk. The total risk value of each plant, available from existing FES analyses, was regressed against the EI for that plant; and average and 95 percent upper confidence bound values of total risk were estimated.

Appendix 2: Valuation of life

Valuing life is a controversial issue, and the estimated values of life vary considerably from one study to another, depending on the way they are calculated. Two approaches have traditionally been used (Viscusi et al, 2000): the first approach estimates the implicit prices for the social risk commodities that may be traded on markets (for instance, workers are willing to accept higher wages for jobs that carry higher risks). The second consists in polling people and ask them how much they value a health outcome. This approach is referred to as the "Willingness to Pay" approach. It is important to remind here that the estimated values are statistical values of life: they do not refer to a specific individual, but rather as the cost to reduce the average number of deaths by one.
Valuing life is needed for certain cost-benefit analyses when health impacts of a regulation have to be monetized. There are many opponents to the use of cost benefit analysis in the environmental regulations, arguing that not only is it impossible to value life but it can also lead to unreasonable results (for instance, smoking should be encouraged based on a cost-benefit analysis since people are expected to die younger and therefore the cost of their retirement on society decreases) (Heinzerling et al, 2000).

In 1997, the USNRC released a document designed to provide guidance for cost-benefit analysis (USNRC, 1997). In that document, the Commission recommended the use of the value $2000/per person-rem averted for both public and occupational exposure, to account for all health effects (and not land contamination). This value was to be used for both routine and accidental exposure. In a paper summarizing a work performed under contract for the U.S. Department of Energy in 1995, Guenther and Thein used a two-fold approach: after estimating the value of statistical life (in dollars), an evaluation of the probability of cancer death due to radiation exposure of some given amount (death per person-Sv) was carried out. The product of both results produced a value per person-Sv. The methods illustrated in the paper are the following:

- *The analysis of jury awards and settlements from wrongful death suits* reflects society's valuation of life thanks to a randomly selected jury. The following table provides a summary of the main assumptions of this method.

Approach: Lawsuit of wrongful deaths			
Method	Advantages	Assumptions	Observations
Examination of jury-awards and settlements in wrongful deaths suits (Otway, 1971; Miller, 1989) from 1989 to 1993 (the time value of money is not taken into account) Distinction between wrongful deaths involving malpractice and wrongful deaths involving product liability	In the case of awards, all aspects (e.g. pain and suffering, loss of service, wrongful deaths and punitive damage) of the award reflect the economic impact of the loss of life	As juries are randomly selected, their decisions "represent a consensus of society's values" The value awarded reflects the value of life remaining.	When the jury specified the remaining years of life, the average annualized awards was 2 to 4 times higher than the calculated average of all the cases There was a small number of large settlements, which skewed the average above the median

Table 7: Valuation of life: Lawsuit of wrongful deaths

- *Another approach is the study of medical expenditures,* which consists in the evaluation of the amount of money "the individual is willing to spend to save or prolong the life of an individual suffering from a debilitating illness". The study carried out chose to analyze cancer.

- *The analysis of insurance coverage* is a third possible method, and assumes that the value individuals place on their own lives is reflected by the amount of coverage they purchase.

- The fourth study carried out consisted in analyzing individual *wages and investments*, which reflect an individual's contribution to society. This approach is very similar to the Human Capital approach.

Finally, the authors performed a literature search for values of life. The results of the various calculation approaches are summarized in Table 8.

Method used to ascertain a value of life	Range in values (1990 U.S. dollars)	Recommended values (1990 U.S. dollars)
Jury award from wrongful death suits	562,000 – 12,760,000	3,454,000
Medical expenditures	141,000 – 4,222,000	4,222,000
Life insurance coverage	130,700 – 3,356,000	3,356,000
Lifetime wages and investments	960,000 – 2,670,000	2,670,000
Review of literature		
500 life-saving interventions	1,297,999 – 191,000,000	2,865,000
Willingness To Pay approach	83,000 – 18,400,000	2,844,000
Human-Capital Approach	210,000 – 1,124,000	558,000
Values used by federal government	2,000,000 – 300,000,000	2,500,000
Law Enforcement		3,017,000
AVERAGE	672,000 – 7,089,000	3,116,000

Table 8: Methods for valuating life

The paper concludes that since the average value of life has been calculated to be $3,116,000; a "conservative" value is $4,000,000 (1990 dollars). The methodologies do not make a difference between early and latent fatalities: for instance, wrongful deaths can be both early and latent. Literature is scarce on how a latent fatality should be weighted in comparison to an early fatality. A value for early fatality five times higher than the value for latent fatality has been used in a societal risk proposal (Okrent, 1981). However, no rationale for such figure is provided. It is reasonable to assume that the statistical value of life for an early fatality is at least as high as that of a latent fatality.

The use of cancer risk factor estimates allow the calculation of the value of a latent fatality: Guenther and Thein estimate the cancer risk factor to be 0.052 Sv^-1; with a range of uncertainty being 0.03 to 0.09 SV^-1. This cancer risk factor value accounts for the fact that the young have an increased sensitivity to radiations, the non-fatal cancers and the severe genetic effects
The value of life has been previously chosen to be equal to 4,000,000 dollars. If the cancer risk factor is estimated to be equal to approximately 0.05 Sv-1; the cost per person-rem is $2000.