The Use of Level 3 PSA in the Risk-Informed, Performance-Based Regulation of Nuclear Power Plants

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REPORT SUMMARY

At the request of the South Texas Electric Generating Station (STP), EPRI assessed the role of probabilistic safety assessment (PSA) Level 3 consequence analysis in the regulation of nuclear power plants. By surveying use of consequence analysis codes, their development, and current status, this report attempts to put Level 3 PSAs in perspective relative to their usefulness for making regulatory decisions.

Background
The Nuclear Regulatory Commission has developed quantitative health objectives (QHO) to ensure public health and safety. While QHOs are not a legal requirement, they do represent NRC's statement of a safety goal policy.

Objectives
- To define the function and describe the structure of Level 3 assessment.
- To summarize the experience with such analyses.
- To provide representative displays of the results.
- To compare available computer codes.
- To provide recommendations concerning the quality and content of PSAs for risk-informed, performance-based regulatory applications.

Approach
Analysts reviewed the structure of actual Level 3 analyses and investigated the impact of different risk metrics. To provide insight on the impact of nuclear plant accidents on public health and safety, they factored dispersal of radioactivity and dose response relationships in humans and land contamination factors into the analysis.

Results
The usual and customary metrics for evaluating risk and safety is based on the likelihood of core damage per reactor year, with a nominal benchmark of 10^{-4} per reactor year. An additional risk metric, large early release frequency (LERF), reflects the role of the containment building. Neither of these metrics were found to directly assess risk to public health and safety. Level 3 analyses, however, provided a better measure of the impact of nuclear facilities based on injury and mortality rates and the consequences of accidents on property damage and restoration costs.

EPRI Perspective

Study results show that Level 3 PSAs use base technology that has been refined over the last 25 years with a broad range of applications. There are a number of computer codes available for Level 3 modeling. Uncertainties in the analysis are mainly driven by the accuracy of Levels 1 and 2 in the models. Provided that Level 1 and 2 portions of the work are realistic and validated, Level 3 results can be sufficiently reliable and dependable for use in making primary regulatory decisions.

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Interest Categories

Risk & reliability
Assessment & optimization

Keywords

Probabilistic safety assessment
Operations
NRC regulations
Licensing
Risk-informed applications

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PERSPECTIVE ON THE ROLE OF LEVEL 3 ANALYSIS

The objective of this report is to place the final phase of a probabilistic safety assessment (PSA) in proper perspective. The final phase is the consequence assessment, oftentimes called the Level 3 model. The recent interest in Level 3 analyses is stimulated by the current pursuit of a risk-informed, performance-based regulatory approach. An industry pilot project is underway to provide an initial feasibility and practicality assessment of this alternative approach to regulating commercial nuclear power plants in the U.S. A brief description of the pilot program is provided in Appendix A.

We intend to put the final PSA in perspective by (1) defining the function and describing the structure of the Level 3 assessment, (2) summarizing the experience with such analyses over the last 20 years, (3) providing representative displays of the results (Appendix B), and (4) comparing and contrasting the various computer codes. We finally identify the conclusions drawn from this experience and the open technical issues in the conduct of a Level 3 analysis.

The original utility-sponsored probabilistic risk (safety) assessments (Oyster Creek, Zion, Indian Point, etc.) were full scope. As such, they encompassed the entire series of events ranging from the initiating event through the scenarios leading to the onset of fuel damage, through core damage and melt, containment failure, release of radioactive material, dispersal of the radioactive material, deposition of the radioactivity and resulting health effects and contamination. These full-scope analyses were divided into three levels for convenience. This segregation process is shown in Figure 1-1.

The structure of the actual analysis is covered in Section 2 of this report.

At present, the usual and customary metrics for evaluating the “risk” or “safety” of a nuclear power plant is primarily based on the likelihood of core damage per reactor-year, with a nominal benchmark of 10^{-4} per year. Sometimes to reflect the role of the containment, an additional risk metric like the likelihood of a “large early release” (LERF) of radioactivity is estimated, without a clear definition of what is large and when is early. The norm for the LERF is 10^{-5} per year. However, neither of these metrics directly assess the risk (or safety) to the health and safety of the public with the operation of a commercial nuclear power plant.

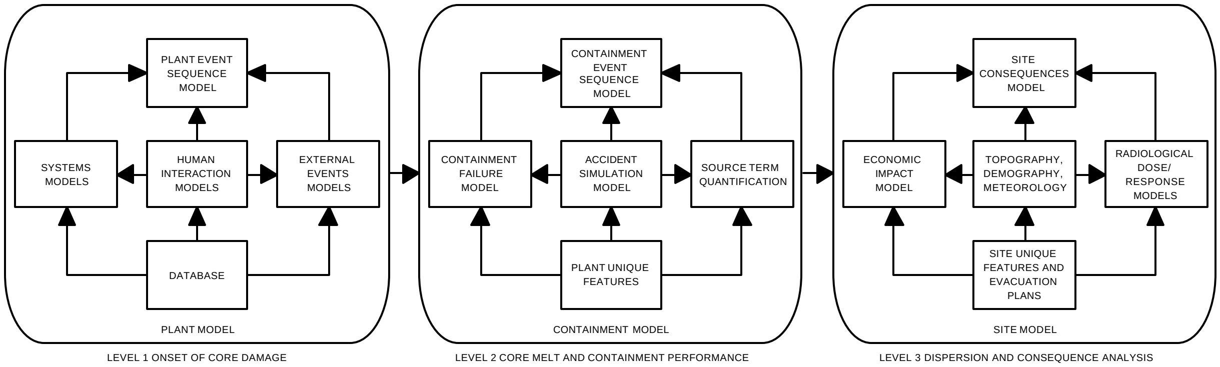

Figure 1-1
Schematic of the Logical Breakdown of a Full-Scope PSA into the Three Levels of Analysis

The U.S. Nuclear Regulatory Commission (NRC) has developed quantitative health objectives (QHO) to ensure that the health and safety of the public is protected. While the QHOs are not law (i.e., required by regulation), they do represent the NRC's statement of safety goal policy. The QHOs are discussed in Appendix C. The policy statement stipulates that the operation of a commercial nuclear power plant should not increase the risk of the public by more than one-tenth of 1% over the "normal" risk to the public without the nuclear power plant. The QHOs account for injuries and fatalities, both prompt and latent.

In a presentation to the Advisory Committee on Reactor Safeguards (ACRS) in 1993, the Director of Nuclear Reactor Regulation of the NRC, Dr. Murley, attempted to correlate the QHOs and the secondary safety indicators, core damage frequency (CDF) and LERF.[1] He concluded that the QHOs could be met with a nuclear plant having a normal containment that had an annual CDF of as high as of 10^{-2}. He concluded that the QHOs and the CDF guideline of 10^{-4} were not in concert. He further stated that the NRC was effectively protecting the investment of the utility, by requiring that the CDF in the secondary risk guideline to be two orders of magnitude lower than it need be to satisfy the QHOs. This dichotomy exists today.

It is also important to recognize that the technology embedded in a Level 3 analysis is neither unique to the full-scope PSA, nor even unique to the nuclear power industry. The same or very similar techniques have been used in the tools used to deal with real (or simulated) nuclear events and releases of toxic and hazardous materials from chemical plants and refineries for more than two decades. More recently, the same techniques are used to respond to terrorist use of chemical and biological weapons and protect troops in the event of their use by hostile forces. These tools have been adequately benchmarked with tests using simulated materials.

We briefly address the issue of uncertainty to help put it in perspective. Often the back end of the PSA is perceived to impart the greatest uncertainty in the overall assessment.

All aspects of a PSA have uncertainties, and all can be treated rationally. For the Level 3 analysis, the input parameters, other than the source terms, are quite well defined. For example, the meteorology of a specific site is well documented, as is the population surrounding the site; and the evacuation routes and times are based on recognized models and approaches. The uncertainty in the radioactive source terms carry over from the Level 2 analyses, and thus are an integral part of the LERF estimation. The uncertainties associated with Level 2 are significant because there is a lack of certainty concerning the phenomenology in the core melt progression and concrete interaction processes. Nevertheless, the uncertainties can be defined and treated properly. There is additional discussion of the Level 3 uncertainties in Section 3.4.5.1 of this report.

Finally, the most important aspect of a full-scope PSA, including Level 3, is the most necessary for making and implementing policy decisions; i.e., how a particular facility or activity affects public health and safety. Once the dispersal of the radioactivity is estimated, the consequence of such dispersals are predicted using well documented dose response relationships and land contamination factors. When assessing the impact of the operation of any engineered facility on the health and safety of the public, the short-term and long-term injury and mortality rates must be assessed. The economic consequences must include property damage and restoration or replacement costs. In many nonnuclear industries, the resulting mortality estimates are euphemistically called "F-N" curves, where F estimates the frequency of N and N is the number of fatalities. In a growing number of European countries, such as the United Kingdom and the Netherlands, the federal government requires industrial facilities to meet quantitative safety goals. The F-N curves are similar to those contained in the WASH-1400 study. One interesting F-N curve is that predicted for the operation of the Trans-Channel railway tunnel, the so-called Chunnel[2] (see Figure 1-2). The increasingly more common use of F-N curves in the transportation, petro-chemical, etc., industries should make the full-scope PSA much more meaningful and useful in the future. A direct comparison between the health and safety impact of Chemical Plant A and Nuclear Plant B becomes possible. The real "risk" of the operation of nuclear plants could then be presented to the public in relatively simple terms for independent evaluation. There is a common misperception of the relative risks in the minds of the general public.[3] The more common risk metrics of CDF and LERF are very abstract and frightful to the many persons without detailed knowledge of the robustness of the design and defense-in-depth licensing basis of nuclear power plants.

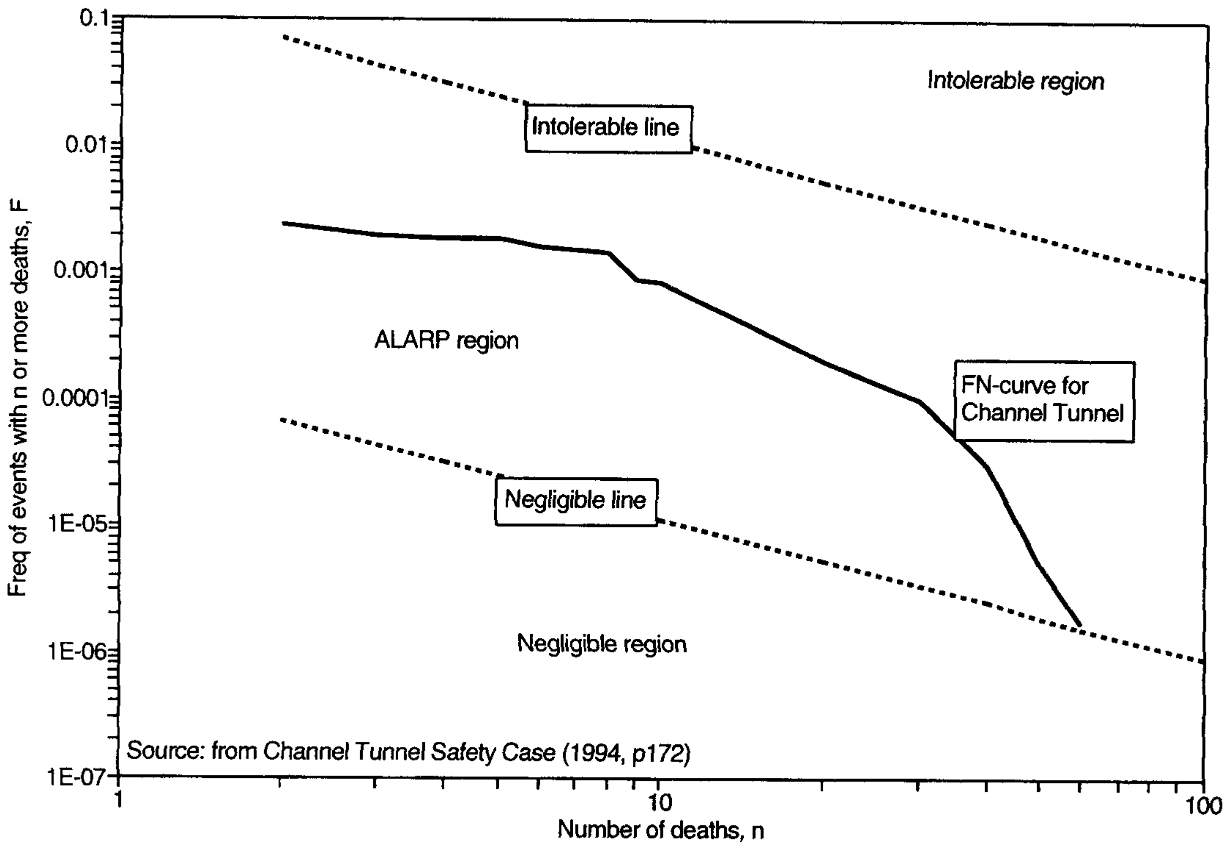

Figure 1-2
Annual Frequency-Number of Fatalities Relationship (F-N Curve) for the Chunnel

2

OVERVIEW OF LEVEL 3 CONSEQUENCE ANALYSIS

2.1 Definition

The Level 3 portion of a PSA is the final step in the calculation sequence. It is intended to incorporate realistic assumptions to analyze the transport and dispersion of radionuclides (defined in the Level 2 analysis) through the environment and assesses the public health and economic consequences of the accident. When combined with the results of Levels 1 and 2, an analysis of this scope permits an assessment of plant risk to the public, since both the consequences and the frequencies of individual or combined accident sequences are estimated (see Figure 2-1).

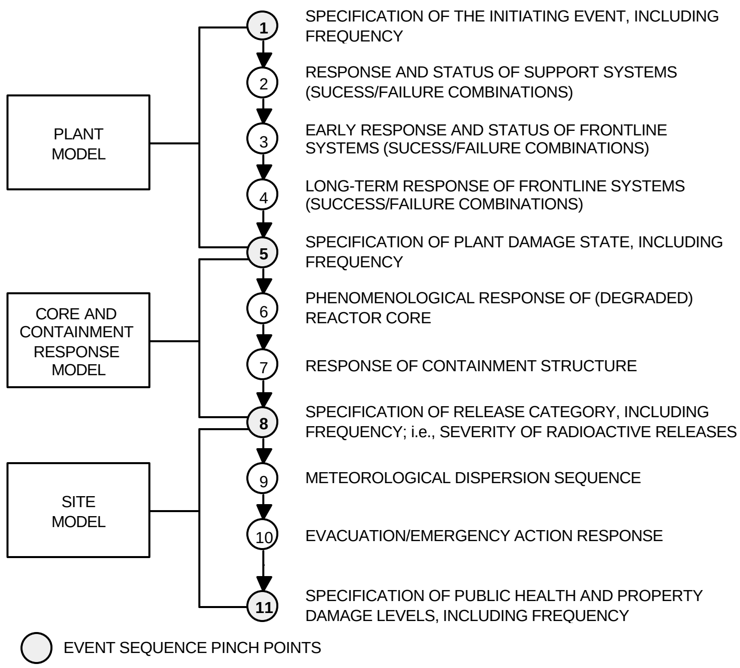

Figure 2-1
Diagram of the Level 1, Level 2, and Level 3 Aspects of a PSA, Showing the Links between the Levels (Indicated as Pinch Points)

Environmental factors can cause the doses to vary substantially for any one release category. To incorporate these variations into the determination of consequences, a given release scenario (category) is simulated repeatedly under many different weather conditions. Other factors, such as population density and evacuations, are treated on a site-specific basis. The results are generally presented in the form of a "risk curve" depicting the frequency of consequences of increasing severity. If uncertainties have been propagated through the three levels, then the results may be expressed as a family of risk curves. Sophistication applied to post processing and report generation would depend on the application. Consequence modeling is intended to incorporate realistic calculations and input assumptions.

2.2 Applications

The practical applications of Level 3 analyses include:

- Evaluations of site specific, societal, or individual risks against the QHOs.
- Support evaluation of new or existing regulatory requirements.
- Presentation of health impacts to the public in a probabilistic manner to support the decision-making process for modifications.
- Addressing issues involving modification of emergency response plans.
- Support of licensing issues involving siting or evaluations against safety goals using standardized designs and reference sites for advanced light water reactors.
- Evaluation of alternative design features, operation and maintenance changes, procedure changes, improvements, or licensing issues.
- Environmental impact assessment (usually not useful for operating plants).
- Accident liability.
- Run, retire, refurbish, replace decisions.

While Level 3 has been very useful in the past for providing input to the regulatory process as regards generic plant siting issues, the focus is now directed toward licensing issues involving plant operations and maintenance. Level 3 results are often surprising, showing small impacts offsite when the offsite consequences had been perceived as significant based on Level 2 results.

2.3 Elements of the Level 3 Analysis

The various elements of a consequence analysis include: (1) transport and diffusion in the atmosphere and/or water; (2) deposition processes [both wet (rain) and dry]; (3) pathways for the accumulation of radiation doses; (4) mitigative measures, such as evacuation or sheltering, that reduce radiation doses; (5) the health effects due to radiation exposure; and (6) economic impacts.

Two basic categories of potential health effects are usually separated into early and chronic (latent) effects. Early effects manifest themselves within days up to a year after exposure. Latent effects (cancers) would probably be observed from 1 year to 40 or 50 years after exposure and genetic effects would occur in later generations.

Early fatalities may result from radiation damage to bone marrow, lung, or the gastrointestinal tract. Radiation damage to the bone marrow is the most important contributor of early effects for light water reactors. Relationships between the radiation dose to critical organs and the probability of fatalities are included in the MACCS code. A number of other health impacts such as prenatal deaths, hypothyroidism, temporary sterility, growth retardation, cataracts, and prodromal vomiting are estimated in the MACCS code. It is not considered to be necessary to address all these effects in detail. Expressing results as predicted numbers of early fatalities and injuries is considered to be sufficient for most Level 3 studies.

Chronic effects include latent cancer fatalities and nonfatal cancers including thyroid nodules. These results are generally presented in Level 3 studies but emphasis in the decision-making process is usually placed on early effects.

Perhaps the most difficult consequence to model is that of economic impact. Property damage from a reactor accident is very different from that resulting from most other catastrophic events. This is primarily because there is no apparent physical damage. The damage arises from radioactive contamination and the possible radiation exposures from continued occupancy. Restrictions on property use results in economic losses that are very difficult to model. Other losses include costs associated with evacuation, agricultural decontamination, and relocation. Compilation of realistic assumptions for the various elements of these costs is very subjective.

2.4 Input Data

The level of detail necessary in a Level 3 analysis will dictate the input data requirements. Each of the required input data types will be addressed along with questions that should be answered when selecting data.

Radioactive source terms for the Level 3 analysis are obtained from each of the release categories generated by the Level 2 analysis. As a minimum, the following source term information is required:

- Core inventory of fission products at the time of reactor shutdown.
- Fractions of the core inventory for each of the risk dominant fission products that is released into the environment. (MACCS can accept a time history of four periods.)
- Heat flux associated with the release of radioactive material for determination of plume rise.
- Timing including time of shutdown, time of start of release, warning time before start of release, delay between warning and start of evacuation, and release duration.

Questions that arise here generally revolve around assumptions to be made for timing. This is very important for emergency planning applications, which directly impacts immediate fatalities. However, emergency planning applications have little impact on latent cancers. Other questions would involve the degree of detail for releases. Generally, if the study requires detailed evacuee doses, detailed release histories are required. General siting studies would require less detail. Level 2 MAAP code output would provide detailed time history data.

Meteorological data used in a Level 3 analysis will vary depending on the analysis being made and availability of data. Parameters include wind speed, wind direction, stability, and rainfall. MACCS uses hourly changes in the atmospheric stability, wind speed, and precipitation for each succeeding hour of travel time. The minimum requirements are:

- Wind speed and direction at heights representative of release height (a single level is sufficient for all but the more detailed studies).
- Stability is expressed as Pasquill-Gifford (PG) stability group.
- Rainfall is a very important parameter since it can rapidly remove particulates from the plume and cause significant doses due to “ground shine.” Many meteorological tower installations do not have these data or the data are of poor quality. A proper Level 3 analysis cannot be done without quality rainfall data. NOAA is a good source of data for any region in the United States.

Terrain height data is not normally required for Level 3 analysis. However, the effect of significant terrain features on wind patterns in the vicinity of the plant must be accounted if risk at specific locations is to be addressed in detail.

Population data include spatial distributions on a radial grid. They are based on census data. Calculations may include both the most current and projected populations. For siting oriented applications, the population distribution need not be made finer than that provided in the census data. For emergency planning applications, greater accuracy in the population distribution and even evacuation time history data are of interest as discussed below. Variables representing population evacuation trajectories and timing will be required to compare evacuation analyses.

2.5 Accident Mitigation

In the U.S., evacuation is generally the preferred mitigation action involving movement of people to avoid or reduce exposure to any hazardous material that may be airborne. The delay time from start of release to start of evacuation has been found to be a very sensitive parameter. This is particularly true for the prediction of early fatalities and injuries, which are very sensitive to the dose accumulated by evacuees from deposited fission products.

Sheltering is also an effective means of reducing external doses and should be considered in all consequence analyses. Consequence codes require shelter (shielding) factors for both cloud shine and ground shine. Different shielding factors are also used for evacuees in transit or stationary and for residents; e.g., in basements at home.

Chronic long-term mitigation actions are also considered in consequence codes. In the U.S., long-term removal of residents from contaminated areas is assumed to reduce the radiation dose accumulated from exposure to the deposited radioactive materials. Mitigation actions in many codes also include interdiction and decontamination. The radioactive contamination will enter foodstuffs like milk, grain, and vegetables. This may require removal (or interdiction) of local consumables from diets. Care must be taken in the use of these chronic models since larger releases do not always result in larger long-term doses due to these interdependencies.

3

SURVEY OF CONSEQUENCE ANALYSIS CODES

3.1 Introduction and Summary

Early in 1994, for an international client, PLG completed the FY-93 survey of the utilization of the MACCS consequences computer code developed by the NRC. The survey showed that while the MACCS code had the capability to perform analyses to support many aspects of the NRC's activities, the more recent applications were in support of U.S. Department of Energy (DOE) applications. Since the application of MACCS and other consequence codes in the U.S. had been somewhat limited, the survey was expanded to include experience with other codes, both in the U.S. and in other countries. This section summaries the results of the augmented survey including reviews of older documents as well as more recent applications.

Since there has been considerable consequence activity in Europe, emphasis has been placed on reviewing European applications. Personal contacts were made with many of the individuals involved in these studies. Communication and display of results were of particular interest in this survey and examples were extracted from the various references and included in this section. This survey also provides the code descriptions including the key features.

3.2 Historical Overview of Consequence Analysis Code Development

Early in the process of licensing the first commercial U.S. nuclear power plants, the U.S. Atomic Energy Commission calculated doses assuming pessimistic meteorology and release of fission products assuming there was no containment. The resulting high doses were reported in the WASH 740 document.[4] When applications for licenses were made for the first commercial power plants with containments in the mid 1960s, the NRC computed doses using 5% probable meteorology and design basis releases assuming an intact containment and limited operation of safety features.

At about the time of the last order for a nuclear plant in the U.S. in 1974, the NRC was conducting the first probabilistic safety study for commercial nuclear plants considering a spectrum of accidents and frequencies. As part of this study referred to as WASH-1400,[5] the first probabilistic dose and health effect consequence code (CRAC) was written. This code was improved by PLG in the 1977-1979 time frame, as part of the first PSA conducted by a utility company. The new code was referred to as

CRACIT. Between 1980 and 1984, CRAC was improved and renamed CRAC2 and a number of new consequence codes (e.g., TIRION, CRACUK, UFOMOD, and ARANO) were written and compared in an international benchmark exercise in 1982.[6]

Based on the experience gained from development and use of the early codes, a number of new, or improved second generation, codes were developed in the 1980s. These included MARC, CONDOR, MACCS, CRACEZ, and COSYMA. As discussed later, MARC (and an updated version called MARC-1) was extensively used by the U.K. in licensing their pressurized water reactors. The CONDOR code has an interesting development history. It was designed as a series of modules over a 4-year period and programmed in only a few months in 1989. As such it has a received a higher level of quality assurance and is conducive to easy replacement of modules as newer ones became available.

Starting in about 1989, the European community pooled their efforts and developed the COSYMA code which has extensive capabilities including improved long-range dispersion and foodchain models. COSYMA is based in large part on UFOMOD and MARC since KfK and NRPB were major contributors. After more than 10 years of development and experience with consequence code applications, another benchmark exercise was conducted in 1993.[7]

More recently, U.S. and CEC development has been focused around MACCS and COSYMA, and a PC version of COSYMA has been released that appears to be very user-friendly. In addition, the DOE has sponsored improvements to MACCS and the latest version called MACCS2 was released in 1995.

Since Chernobyl, there has been greater interest in Europe on long-range transport, foodchain, and food ban analytical models than in the U.S. This is reflected in the code capabilities, where COSYMA has greater capabilities than MACCS in these areas. It appears that at the present time, most CEC funding is directed toward a real-time consequence model called RODOS.

The time line in Figure 3-1 helps put this code development history into perspective. It also serves as a reference for some of the discussions that follow.

3.3 Scope of Study

The scope of work for this study was separated into four phases including information gathering, interviews, documentation review, and evaluation. The primary interest was on airborne and terrestrial pathways; however, some interesting information concerning liquid (aquatic) pathways was found as discussed in Section 3.8.

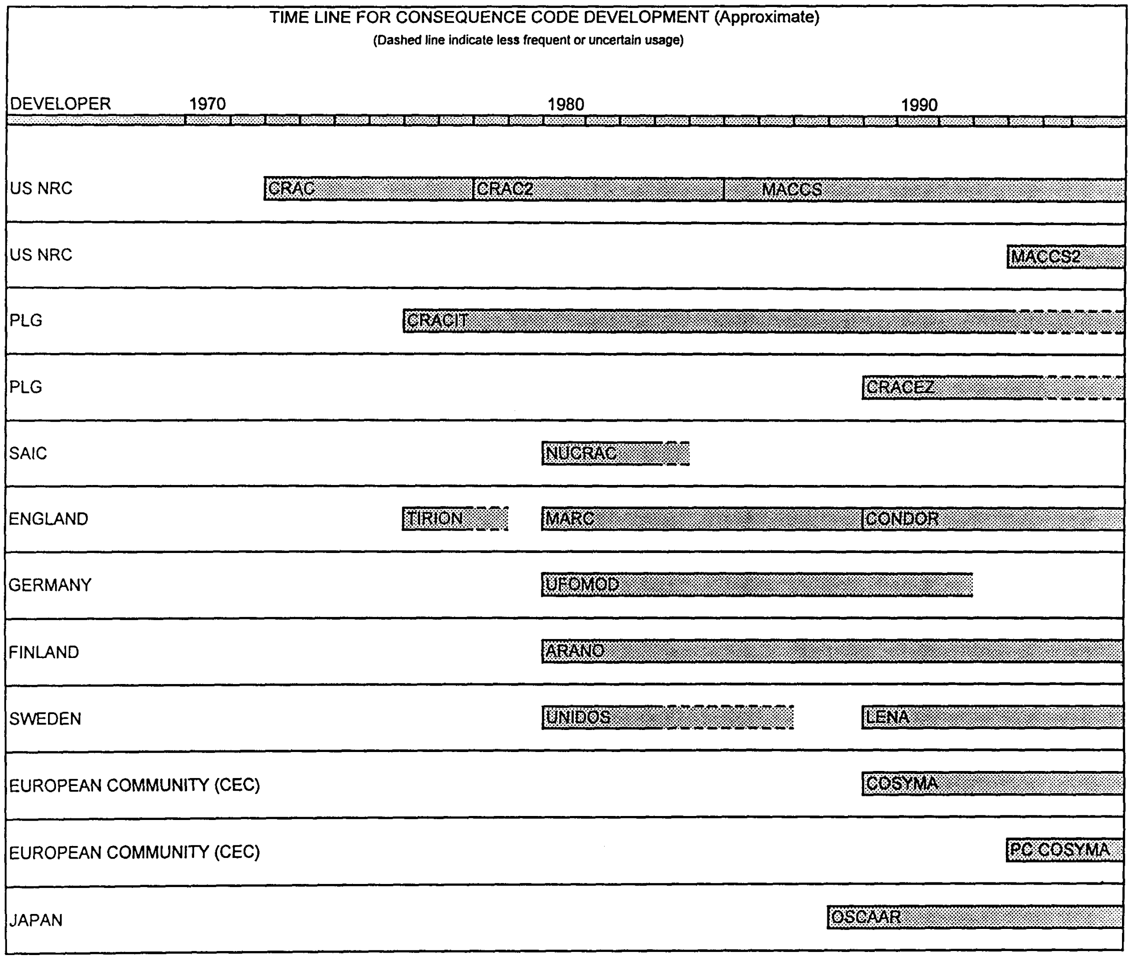

Figure 3-1
Time Line for Consequence Code Development (Approximate)

3.3.1 Information Gathering

The work scope involved a literature search in conjunction with discussions with principal analysts involved in the studies where appropriate. Each document that was relevant to this survey was reviewed and evaluated in each of a series of evaluation categories given in Section 3.3.4. More than 100 documents were reviewed and 43 were selected for evaluation in this study.

3.3.1.1 Document Search in U.S.

During the initial phase of the survey, an information search for the following information was conducted in the U.S.:

- Contact DOE laboratories and find publicly available reports.

- Contact NRC to obtain updated assessment of activities (cost benefit, siting, emergency planning).
- Search for more utility applications and obtain reports.
- Obtain lists of MACCS users from code center and users groups.
- Ascertain availability of the three ALWR studies.
- Find any modifications to existing codes or new codes under development at the national labs or elsewhere.
- Obtain latest status of MACCS2.

3.3.1.2 Document Search Outside U.S.

In parallel with the U.S. search, the following contacts were made to locate documents outside the U.S.:

- In Germany, UFOMOD authors at KfK were questioned to determine applications and find available results and reports.
- In Holland, representatives of KEMA were contacted to find emergency planning applications.
- In the U.K., representatives of AEA technologies and NRPB were contacted to find results of site selection and other more recent studies.
- Also, in the U.K. documents describing use of the MARC code for the Sizewell and Hinkley public inquiry process were requested.[8]
- European community representatives and attendees at the Level 3 Technical Committee meeting in November 1994 in Vienna were questioned regarding recent applications.

3.3.2 Discussions with Code Developers/Users

To obtain a more thorough understanding of consequence analysis activities, the literature search was supplemented by a series of discussions with persons active in development and utilization of consequence codes. These discussions were useful in finding additional references, clarifying code capabilities, and determining the acceptability of the various applications. There is concern that consequence analyses are not understood by the decision makers who continuously ask for the “worst case” results.

3.3.3 Review of Documentation

To proceed in an orderly manner, a method of identifying documents was developed and followed throughout this study. In addition, questionnaires were used to assist in addressing the evaluation topics in the document reviews.

3.3.3.1 Document Identification System

Each document was classified into one of 12 categories and given a PLG identification number. Those in the first 10 categories received more thorough evaluation and those in the last 2 categories were used to determine consequence code capabilities. The document identification system is outlined in Table 3-1.

When reference is made in the text to one of these documents, the two category letters will appear followed by a number; e.g., UC-3 is the third document in the U.S. Cost-Benefit category.

Table 3-2 provides a list of documents (with document identification) considered appropriate for more detailed evaluation in this survey.

3.3.4 Evaluation Topics

For each document that was determined to be of value in this survey, a series of evaluation topics were addressed and documented. Discussions in Section 3.4 and 3.5 are based on the document evaluations. These topics are as follows:

- **Purpose.** What was the purpose of the study?
- **Results.** What were the results?
- **Source Term.** What was the basis for selecting the source terms used?
- **Weather Data.** – What was the source of input weather conditions?
- **Pathway and Health Effects.** Which dose pathways and health effects were examined?
- **Countermeasures.** Were protective actions assumed? What were the criteria?
- **Output.** What was the output format, especially with regard to supporting the decision-making process? Appendix B provides examples of consequence analysis results copied from the various documents.

Table 3-1
Document Identification System

Identifier	Topic
UC	U.S. Cost-Benefit and Other NRC Activities
US	U.S. Site Selection
UA	ALWR Design and Licensing
UD	DOE Facilities
UO	Other Applications
EG	German (and Other European)
EH	Holland
EE	UK-General
EI	UK-Hinkley and Sizewell Inquiry
EO	Other Applications
MC	MACCS Code
OC	Other Codes

Table 3-2
List of Documents Reviewed

UC — U.S. Cost Benefit and Other NRC Activities

UC-1 "Re-Evaluation of Policy Regarding Use of Potassium Iodide (KI) After a Severe Accident at a Nuclear Power Plant," 1989

UC-2 "Staff Approach for Assessing the Effectiveness of the Present Regulations with Respect to the Commission's Safety Goals," 1993

UC-3 "Assessment of ISLOCA Risk--Methodology and Application to a Combustion Engineering Plant," NUREG/CR-5745, EGG-2650, 1992

UC-4 "Core Melt Accident Dose-Versus-Distance Probability Distributions 25% Power Operation," Shoreham Nuclear Power Station, PLG-0542, 1987

UC-5 "Transmittal of Results of INEL Review of LILCO Offsite Consequence Analyses," (FIN D6023), Oben-84-87, 1987

UC-6 "Transmittal of Letter Report Documenting Consequence Calculations for Shoreham 25% Power Operations Using Modified Source Terms," RJD-16-88, 1988

US — U.S. Site Selection

US-1 "Technical Guidance for Siting Criteria Development," NUREG/CR-2239, SAND81-1549, 1982

US-2 "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, 1990

UA — ALWR Design and Licensing

No documents have been released to the public--all are proprietary

UD — DOE Facilities

UD-1 "Consequence Assessment for the High-Level Waste Tanks Probabilistic Risk Assessment," 1994

UO — Other Applications

UO-1 "Lessons Learned from an Uncertainty Analysis of Plume Dispersion," 1994

UO-2 "Seabrook Station Risk Management and Emergency Planning Study," PLG-0432, 1985

UO-3 "Indian Point Probabilistic Safety Study," Power Authority of the State of New York, 1982

UO-4 "Display of Risk Reduction Due to Countermeasure Strategies in Consequence Models," K. Woodard, 1994

EG — European - German

EG-1 "Differences in the Radiological Effects of a Major Accident Using Heu or Leu Fuel Elements at the BER II," 1993

EG-2 "On the Confinement of the Radiological Source Term during Beyond Design Basis Events in Future Pressurized Water Reactors," 1994

EG-3 "Illustrative Applications of Accident Consequence Assessment Codes," 1990

EG-4 "Notfallschutz und Vorsorgemabnahmen bei kern-technischen Unfallen," 1991

EG-5 "Zur Eingrenzung des radiologischen Quellterms bei auslegungsuberschreitenden Ereignissen in zukunftigen Druckwasserreaktoren," KfK 5199, 1993

EG-6 "Konsequenzen und Wirksamkeit von Umsiedlungsmabnahmen nach kertechnischen Unfallen," KfK 4990, 1992

EH — Holland

EH-1 "UFOMOD/COSYMA Consequence Calculations in the Netherlands," 1990

EH-2 "Milieu-Effectrapport Modificaties Kernenergie-Eenheid Centrale Borssele," 1993

EH-3 "Wijzigingen Kernenergiecentrale Dodewaard," 1994

EE — English - General

EE-l "The Application of Accident Consequence Analyses in Siting Studies," 1985

EE-2 "Workshop on Methods for Assessing the Off-Site Radiological Consequences of Nuclear Accidents," Proceedings 15-19 April 1985, Luxembourg, PB87-209664, 1986

EE-3 "The Influence of Season of the Year on the Predicted Agricultural Consequences of Accidental Releases of Radionuclides to Atmosphere," NRPB-R178, 1985

EE-4 "Verification and Validation of NRPB Models for Calculating Rates of Radionuclide Transfer through the Environment," NRPB-R223, 1989

EE-5 "COCO-1: Model for Assessing the Cost of Offsite Consequences of Accidental Releases of Radioactivity," NRPB-R243, 1991

EI — English - Hinkley and Sizewell

EI-1 "An Assessment of the Radiological Consequences of Releases from Degraded Core Accidents for a Proposed PWR at Hinkley Point," NRPB-M141, 1987

EI-2 "Assessment of the Radiological Consequences of Releases from Degraded Core Accidents for a Proposed PWR at Hinkley Point: Results Using MARC-1," NRPB-M152, 1988

EI-3 “Assessment of the Radiological Consequences of Releases from Design Basis Accidents from a Proposed PWR at Hinkley Point, Using MARC-1,” NRPB-M153, 1988

EI-4 “Assessment of the Radiological Consequences of Releases from Containment Bypass Accidents from a Proposed PWR at Hinkley Point, Using MARC-1,” NRPB-M154, 1988

EI-5 “Agricultural Consequences of Accidental Releases from a Proposed PWR at Hinkley Point: The Effects of Assuming that EC Regulations on Food Intervention Levels are Applied,” NRPB-M155, 1988

EI-6 “The Influence of Countermeasures on the Predicted Consequences of Degraded Core Accidents for the Sizewell PWR,” NRPB-R163, 1983

EI-7 “The Radiological Consequences of Degraded Core Accidents for the Sizewell PWR: The Impact of Adopting Revised Frequencies of Occurrence,” NRPB-R160, 1983

EI-8 “The Radiological Impact on the Greater London Population of Postulated Accidental Releases from the Sizewell PWR,” NRPB-R146, 1983

EI-9 “Degraded Core Accidents for the Sizewell PWR: A Sensitivity Analysis of the Radiological Consequences,” NRPB-R142, 1982

EO — Other Applications

EO-1 “Procedures for Conducting Independent Peer Reviews of Probabilistic Safety Assessments,” 1994

EO-2 “Probabilistic Consequence Analysis of Research Reactor Accidents - Comparing the Predictions of Different Consequence Assessment Codes,” 1994

EO-3 “MACCS Calculations for the VVER-Type Reactor at the Paks NPP (Hungary),” 1994

EO-4 "HERALD - A Program for Analyses of Radiation Accident Consequences," abstract of a paper to Technical Committee Meeting on Procedures and Computer Codes for Level-3 Probabilistic Safety Assessment, 1994

EO-5 "The Experience and Expectations from the Probabilistic Consequence Analysis (PCA) Codes," 1994

EO-6 "Notes to Problems of Level-3 PSA Calculations," 1994

_______EO-7 "Radioactive Contamination of an Urban Environment Under Winter Conditions," 1994

MC — MACCS (and CRAC2)

MC-l "INEL Personal Computer Version of MACCS 1.5," NUREG/CR-5667, EGG-2634, 1991

MC-2 "Joint CEC/USNRC Uncertainty Study: Dispersion and Deposition Panel," 1994

MC-3 "A Review of the MELCOR Accident Consequence Code System (MACCS): Capabilities and Applications," 1994

MC-4 "Difficulties Encountered When Attempting to Use MACCS in an Arctic Area," 1994

MC-5 "Comparison of MACCS Users Calculations for the International Comparison Exercise on Probabilistic Accident Consequence Assessment Codes," NUREG/CR-6053, BNL-NUREG-52380, 1994

MC-6 "Use of Post-Chernobyl Data from Norway to Validate the Long-Term Exposure Pathway Models in the Accident Consequence Code MACCS," 1993

MC-7 "MACCS2: An Improved Code for Assessing Nuclear Accident Consequences"

_______MC-8 "A Comparison of Uncertainty and Sensitivity Analysis Techniques for Computer Models," NUREG/CR-3904, SAND84-1461, 1985

MC-9 "Sensitivity and Uncertainty Studies of the CRAC2 Computer Code," NUREG/CR-4038, ORNL-6114, 1985

OC — Other Codes

OC-1 "CONDOR 1: A Probabilistic Consequence Assessment Code Applicable to Releases of Radionuclides to the Atmosphere," SRD R598, TD/ETB/REP/7021, NRPB-R258, 1993

OC-2 "Probabilistic Accident Consequence Assessment Codes," Second International Comparison, Overview Report, 1994

OC — Other Codes (continued)

OC-3 "Probabilistic Accident Consequence Assessment Codes," Second International Comparison, Technical Report, 1994

OC-4 "COSYMA A New Programme Package for Accident Consequence Assessment"

OC-5 "COSYMA: Users Intercomparison Exercise"

OC-6 "PC COSYMA: An Accident Consequence Assessment Package for Use on a PC," 1994

OC-7 "Proceedings of the First COSYMA Users Group Meeting 25-26 April 1994 Arnhem, The Netherlands"

OC-8 "PC COSYMA PC Version of the Probabilistic Accident Consequence Code COSYMA," 1993

OC-9 "COSYMA Users Intercomparison Exercise," 1994

OC-10 "COSYMA, A Mainframe and PC Program Package for Accident Consequence Assessments"

OC-11 "CONDOR: The UK Probabilistic Accident Consequence Code," 1994

OC-12 "NRPB Methodology for Assessing the Radiological Consequences of Accidental Releases of Radionuclides to Atmosphere--MARC-1," NRPB-R224, 1988

OC-13 "Uncertainty Analyses of the Countermeasures Module of the Program System UFOMOD," KfK 4472, 1989

OC-14 "Procedures for Uncertainty Analyses of UFOMOD A User Guide," KfK 4626, 1990

OC-15 "Uncertainty Analysis of the Food Chain and Atmospheric Dispersion Modules of MARC," NRPB-R184, 1988

OC-16 "Comparison of the MARC and CRAC2 Programs for Assessing the Radiological Consequences of Accidental Releases of Radioactive Material," NRPB-R149, 1983

- **Problems.** Were any issues or problems with the consequence code identified?
- **Appropriateness of MACCS.** Is there any obvious reason why MACCS would not have been appropriate for this application?

3.4 Applications in the U.S.

The most popular consequence code used for all purposes in the U.S. is the MACCS code and CRAC/CRAC2 before that. However, for full-scope PRAs performed by utilities, PLG's CRACIT code was used most often. Since very few, if any, Level 3 PRAs are currently in progress for nuclear power plants, the scope of this survey is directed more toward applications of all types—not just power plants (although some PRA activities for power plants are discussed). Applications included are cost-benefit (backfit for system improvement), site selection, advanced designs (ALWR), DOE facilities, emergency planning, international comparisons, source term determination and uncertainty analyses. These applications will be addressed below, where only brief comments are presented on certain aspects of the evaluations

3.4.1 Cost-Benefit and Other NRC Activities

Limited cost-benefit and other NRC activity in the U.S. was identified, and evaluated in this section as discussed below. The documents reviewed are identified in Table 3-2 under the "UC" category.

3.4.1.1 Use of Potassium Iodide (KI)

Over the past 20 years, there has been considerable discussion concerning the effectiveness of using KI in emergencies. The NRC has sponsored several studies and in the document (UC-1) presents a simplified cost-benefit calculation. The cost versus benefit ratio was 54 and, thus, the use if KI was not considered to be cost beneficial. Results were based on WASH-1400 assumptions and there are no graphs or tables of interest to the display of results. It is presented here as a rather interesting example of the simplified use of PRA Level 3 calculations.

3.4.1.2 Safety Goals

The NRC has been developing safety goals and providing bases in Level 3 terms for many years. While there is no analysis presented in the UC-2 document, it does provide an example of the thought process and use of NUREG-1150 results to develop regulatory policy. One of the main objectives of regulators is to know what the risks are for the existing plant/site characteristics before establishing new criteria (or safety goals). One would want to set goals that are reasonably achievable. The NUREG-1150 results for several U.S. plants provided additional benchmark information.

3.4.1.3 Interfacing System LOCA (ISLOCA)

A very good example of the use of MACCS for a Level 3 analysis is found in the UC-3 document. Here, the NRC has used its subcontractor INEL to perform a Level 3 analysis to demonstrate the sensitivity to decontamination factors typically assumed in ISLOCA evaluations. The display of results is unique, in that mean fatalities were plotted as a function of decontamination factor, see Figure B1.

3.4.1.4 Emergency Planning at Shoreham

Three documents were evaluated (UC-4, UC-5, and UC-6) that were submitted several years ago to justify operation of the Shoreham plant at reduced power level such that the typical emergency response preparations would only be required less than 1 mile from the plant. This was done using the CRACIT and CRAC2 codes to compute dose versus distance versus conditional (on core melt) probability curves (see Figure B2). These curves were compared with the NUREG-0396[9] curves used to establish the original 10-mile EPZ for planning that is required for all nuclear plants in the U.S. The comparison clearly showed that the 200 rem curve that had a “knee” at about 10 miles would have a knee at less than 1 mile if Shoreham operated at 25% power. This is an excellent example of the use of Level 3 for emergency planning.

The UC-5 document was prepared by an independent NRC subcontractor who provided an independent check of PLG’s results in UC-4 and found that they were in close agreement.

The UC-6 document has a unique presentation of results pertinent to emergency evacuation. Plots are given of probability of dose at two miles versus time after start of release. This allows comparison of evacuation requirements for 25% and 100% power and shows the extra margin in terms of time before evacuation is required to avoid a 200 rem dose for the 25% power case (see Figures B3 and B4).

At least two other applications have been made in the U.S. where Level 3 has been used to support emergency planning. The documents both compared the same dose versus distance plot with the NUREG-0396 plot. In the Seabrook case (see document UO-2), other displays were used and are discussed in Section 3.4.5. The San Onofre case used Level 3 in public hearings to justify the use of a 10-mile EPZ in California. Comparisons of dose versus distance with NUREG-0396 curves were also used; however, documents for this hearing could not be located.

3.4.1.5 Backfit

The NRC allows reactor operators in the U.S. to apply for permission to make changes to systems and procedures. As part of the evaluation process, the costs and benefits of making the change may be considered. Typically, those calculations use a “cookbook” approach rather than make independent and site/plant-specific calculations with

MACCS or some other code. When the cookbook approach is used the population dose (person-rem) values for various damage scenarios are taken from NUREG-0933[10] using values based on WASH-1400. The monetary cost is usually valued at $10,000 per person-rem.

Recently, a utility has used MACCS to justify changes in the containment leak rate test frequency. The MACCS calculations were done with the design containment leak rate and then with a degraded leak rate. The difference in mean person rem for the two cases was very small, justifying the proposed change in procedures. Other utilities may make similar risk-based proposals in the near future. Copies of submittals to NRC were not available for this study. When reviewing these submittals, the NRC apparently does not perform independent consequence analyses using computer codes.

3.4.2 Site Selection

Two major NRC uses of Level 3 analysis are presented here that fall into the "US" document category.

3.4.2.1 Reactor Siting Study

A major study related to site selection was conducted by Sandia for NRC in 1982 (see US-1 document). A more detailed summary follows.

3.4.2.1.1 Purpose

Its major purposes were (1) to develop new technical guidance to support the formulation of new regulations for siting nuclear power reactors, and (2) to take existing site characteristics into account and determine the consequences of possible plant accidents, and determine socioeconomic impacts. Guidance was requested regarding criteria for population density and distribution surrounding future sites, and standoff distances of plants from offsite hazards.

3.4.2.1.2 Results

The results of the study relating to the consequences of possible plant accidents can be summarized as follows.

- All consequence calculations were performed with the CRAC2 code.

- Estimates of the number of early fatalities are very sensitive to source term magnitude. The mean number of early fatalities decreases about two orders of magnitude for a one-order-of-magnitude decrease in the source term. The source terms used in the study neglect or underestimate important fission product retention mechanisms in the primary coolant system and containment, so the magnitudes of consequences may be overestimated significantly.

- The weather conditions at the time of a large release have a substantial impact on the number of health effects caused by the release. In contrast to this, it is stated in the document that mean health effects (i.e., health effects averaged over many weather sequences) are not very sensitive to meteorology.
- Peak early fatalities generally are caused by ground shine after washout (rainout) of particulates in the plume onto the ground in a populated area.
- The distance out to which consequences might occur depends principally upon source term magnitude and meteorology. The maximum calculated distances are associated with improbable events, such as rainout of the plume onto a population center.
- Calculated consequences are very sensitive to population distribution. Mean results generally are determined by the average density of the entire exposed population, while peak results are determined by the distance to and size of the exposed population centers after significant travel without rain (delayed rain).
- Early fatalities can be reduced significantly by emergency response actions (such as sheltering or evacuation).

3.4.2.1.3 Source Term

The study used five NRC-defined siting source terms (named SST1 through SST5). These source terms were chosen to represent a variety of possible accidents, ranging in severity from extensive core damage with early containment failure to limited core damage in an essentially intact containment. The most severe source term involved release of 100% of fission product noble gases, 45% of radioiodine, and two-thirds of the cesium and rubidium inventory. In contrast, the least severe source term released almost no fission product noble gases or particulates.

The fission product core inventory was estimated for a 3,412 MWth (~ 1,120 Mwe) reactor with an end-of-cycle fuel burnup of 33,000 MWd/MTU. This inventory was scaled linearly for other size reactors evaluated in the study. These calculations were performed with the SANDIA-ORIGEN computer code.

3.4.2.1.4 Weather Data

The CRAC2 code used to estimate consequences in the study requires 1 year of contiguous hourly data, which are sampled statistically. Meteorological input consisted of hourly recordings of weather conditions from 29 National Weather Service stations in the U.S., along with seasonal atmospheric mixing heights. Site wind roses were obtained from Environmental Impact Reports and Safety Analysis Reports for the sites under study.

3.4.2.1.5 Pathway and Health Effects

CRAC2 models dose pathways to man that include exposure from airborne radionuclides in the plume (cloudshine), exposure from radionuclides deposited onto the ground (ground shine), and internal doses from inhalation of radionuclides, and from ingestion of contaminated food. The principal health effects are early fatality and injury and latent cancer.

3.4.1.2.6 Countermeasures

Consequence calculations were performed for a variety of emergency response scenarios, ranging from prompt evacuation, to sheltering, to no emergency response at all. The evacuation model in CRAC2 is relatively simple since it models radial travel of the population at a given speed.

3.4.1.2.7 Output

The output was in the form of means for each source term and CCDFs for each site for the SSTI source term. Groups of sites were included on each CCDF plot for early fatality, injury and latent cancer.

3.4.1.2.8 Problems

The major problem that occurred was with the interpretation by the news media of the "tails" of the CCDFs for the SST1 source term. The CCDFs showed many thousands of fatalities at low frequency of occurrence for many sites. There is no real solution for this problem except perhaps to indicate the frequencies of every day risks taken by individuals (e.g., aircraft crash, etc.) to add some perspective. This is difficult to do in the CCDF format.

3.4.2.2 Severe Accident Risk Study, NUREG-1150

After many years of preparation, NRC published NUREG-1150. It relied heavily on the MACCS code and included uncertainties. Excerpts are included from NUREG-1150 in the US-2 document. Since the report is well known, this study discusses only the presentation of results. A large part of the results are in tabular form (Figures B5 through B8) which are supplemented by several interesting graphical displays. The typical CCDFs are presented as a family of curves in Figure B9. Figures B10 and B11 use bar graphs and pie charts to provide intermediate results. The "clipped" bars in

Figures B12 and B13 effectively compare the five reactors evaluated and the bar charts in Figures B14 and B15 are used to assess the relative effectiveness of emergency response actions. The bar chart in Figure B16 shows the difference in mean fatalities due to following different protective action strategies.

3.4.3 ALWR Licensing Review

Under the licensing provisions of 10 CFR Part 52, applicants may submit design documentation for certification of standardized nuclear designs. Two U.S. Nuclear Steam Supply (NSSS) vendors have submitted and obtained design certification approval for evolutionary advanced light water reactor (ALWR) (GE – ABWR, ABB C-E – System80+) designs. In addition, a third, passive ALWR design (Westinghouse – AP600) has been submitted and is in the approval process.

The development of ALWR designs was spearheaded by U.S. utilities, with the participation and support of several international utilities and the close cooperation of the DOE. These organizations established the technical foundation for future, standardized ALWR designs; a set of utility design requirements. The product of this activity, which was managed by EPRI, is the Utility Requirements Document (URD).

The URD requires that ALWR designers calculate off-site consequences from severe accidents (Level 3 analysis) at an ALWR reference site, which conservatively represents the consequences of most potential domestic sites. The URD requires the designers to demonstrate that the cumulative frequency is less than 1×10^{-6} per reactor-year for sequences resulting in greater than 25 rem whole body dose over 24 hours at 0.5 miles from the reactor. In addition, plants which are being designed to meet the ALWR emergency planning criteria are to demonstrate a cumulative frequency of less than 10^{-6} per reactor-year for sequences resulting in greater than 1 rem (TEDE) over 24 hours at the site boundary. The offsite consequences calculations for ALWRs are performed using the MACCS, or CRAC2 computer codes.

3.4.4 DOE Analyses

Most activity using the MACCS code in the U.S. has been in support of DOE facilities. As discussed in Section 3.7, changes to the MACCS code were deemed necessary by DOE and a new version called MACCS2 was developed. In addition, another unofficial version was developed at Rocky Flats that includes the effects of building wake and other factors necessary for computing onsite doses. This version is called MACCS2W and is not available outside DOE.

The following section presents a limited review of various DOE applications in the "UD" document category, including use of codes other than MACCS.

3.4.4.1 Waste Tank Risk

Los Alamos National Laboratory has conducted a probabilistic safety analysis for the high level waste storage tanks at Hanford for DOE. In this assessment (see document UD-1), a code referred to as AP-RISK was used. It was selected because the release scenarios involved large particulate releases with a broad particle size distribution. MACCS does not have the capability to easily take account of different particle sizes in a single run. While the AP-RISK code did not use sequential meteorological data, lid height or rain data, it was considered more suitable than MACCS since more accurate treatment of dry deposition was deemed more important. Results were expressed as CCDFs for population dose and economic consequences (see Figures B17 and B18).

3.4.4.2 Other DOE Studies

In discussions with DOE representatives, it was learned that other studies have been accomplished using MACCS but no reports are available. These studies were conducted for the Savannah River facility, Rocky Flats, INEL, and Sandia. Some of these applications required modifications to treat isotopes that were not of concern for WASH-1400 type reactor studies. This was a major reason for changes leading to development of MACCS2.

In most cost-benefit analyses used to justify changes to procedures or equipment, it is argued that rigorous consequence assessments are unnecessary due to the very low frequency of the events leading to high consequences.

3.4.5 Other U.S. Activities

In Sections 3.4.1 through 3.4.4, the newer uses of consequence analysis in the U.S. were discussed. Since this study has a strong emphasis on presentation and display of results, several additional (older) documents more related to previous power plant PRAs and emergency planning applications are discussed here. Issues such as uncertainty, public perception of risk and initiator identification are of note in the “UO” document category.

3.4.5.1 Uncertainty

Probably the most difficult topic to address in a Level 3 analysis is uncertainty. A typical Level 3 calculation involves more than a dozen variables, all of which have their own uncertainty distribution. Many attempts have been made to quantify uncertainty with little practical success. Some of the better studies are reported in Iman[11] and in the MC-2, MC-8 and MC-9 documents.

In a paper recently at an ANS meeting (see UO-1), the NRC (and CEC) reported an attempt to quantify the uncertainty distribution in the MACCS atmospheric dispersion model. The authors expressed frustrations in the accomplishment of this task. The original plan called for establishment of five additional expert panels. It is not clear that this effort will continue. Additional papers have been presented since the ANS meeting that include more detail. The overhead projector sheets in document MC-2 represent a very detailed summary of the uncertainty quantification for atmospheric dispersion parameters. These were presented at the IAEA technical meeting in November 1994 by Ms. Mary Young of Sandia. Both MACCS and COSYMA utilize the Gaussian plume model. Therefore, the panel of experts were requested to express their opinions based on measurable data on the 5th, median, and 95th percentile for plume concentrations and dimensions at various distances downwind. Upper and lower bounds were requested but not required. The experts were briefed on the probabilistic aspects of how their information would be used and each performed his function independently of others in the group. The results were processed (aggregated) to obtain a final expression of uncertainty to compare with point estimates used in the computer models. The final iteration included 101 dispersion, 70 dry deposition and 36 wet deposition cases.

One interesting aspect was use of "seed" variables known to the staff but not the experts. These were based on field tests and were used to provide feedback during training and to evaluate performance of the expert.

Results are presented as three-dimensional bar graphs in the MC-2 document to effectively show differences between experts. Results show that the experts were in general agreement and chose values that were somewhat different than the values used in the consequence codes. The conclusions reached were as follows:

- Distributions on deposition and dispersion consequence code input parameters emulate the distributions on the elicited deposition parameters.
- Performing uncertainty analysis that goes beyond parameter uncertainty using fixed codes presents difficulties.
- Distributions developed have many research and regulatory applications beyond this study.
- The aggregated distributions are model independent and to some extent address model uncertainty.
- Distributions include state of knowledge uncertainty (uncertainty in initial conditions) and stochastic uncertainty.
- Uncertainty analyses can be performed on dispersion and deposition modules of MACCS and COSYMA.

The IAEA paper (MC-2 document) was more optimistic than the ANS paper (UO-1) as to the usefulness of the techniques developed. However, both conclude that the exercise provided useful insights of value to further attempts at quantifying uncertainties.

3.4.5.2 Emergency Planning at Seabrook

In the mid 1980s, PLG performed a number of analyses to support emergency planning at Seabrook. The report (document UO-2) is mentioned in this study because it has a series of unique displays of consequence results. The tabular display in Figure B19 is used to evaluate the effectiveness of a plant procedure change. Figures B20 and B21 use CCDFs on one page to show the effectiveness of evacuation distance. Risk versus distance curves are presented in Figure B22. Risk reduction due to evacuation out to a certain distance is shown in Figure B23. The spatial distribution of mean fatalities is plotted without protective actions in the Figure B24 example and Figure B25 shows the fatality risk as a function of evacuation distance. Safety goals are compared in bar chart format on Figure B26. This suite of graphs represents the most complete display of consequence results found to support a licensing request.

3.4.5.3 Family of Curves for Indian Point PRA

During the 1980s, PLG performed a number of full-scope PRAs that included Level 3 results. In each case, uncertainties were computed and displayed in various formats. A few examples are discussed below. Typically, each of the normal Level 2 release categories was separated into four groups with different source terms and timing characteristics to express uncertainties in the release category. Each of these was run through three different consequence code passes with different input parameters, representing uncertainties in the consequence model. Results were then frequency weighted and combined using a convolution technique to obtain the “family of curves.”

In a study which is part of an early PSA for Indian Point (document UO-2), PLG used the “family of curves” concept which provided CCDFs with varying probability levels (see Figures B27 through B29). Also, probability density curves were used to express uncertainty and show which initiators were important contributors to health effects (Figure B30). It is believed that this presentation may benefit some reviewers in understanding the magnitude of uncertainties. A bar chart format was used to show the relative importance of plant damage states to more than 100 early fatalities in Figure B31. These displays were used in public hearings during the early 1980s to justify continued operation of the plant. Apparently, they were well received by the hearing board.

3.4.5.4 Risk Map

Finally, as part of code development in recent years, PLG has developed the “risk map” concept (see document UO-4) that displays color-coded locations on a site map where risk is higher and lower. It also provides a convenient format for showing how mitigative action, such as evacuation, reduces risk in affected areas (see Figure B32). This format may be easier for the public to understand since they can perhaps visualize the situation spatially (on maps) more effectively.

3.5 Applications in Europe

Most of the preceding discussion concerning U.S. applications has centered around activities near the site. In Europe, the focus has shifted after the Chernobyl accident such that long-range transport and foodchain exposure has become far more important in influencing the direction of their consequence code development.

The early activity in Europe in the 1980s involved development of a series of codes generally following the CRAC code. Codes like MARC in the UK were used extensively to support licensing of their newer plants. The first benchmark exercise helped conform the analyses and served as a quality assurance check.

The following discussion evaluates both older and more recent uses of the codes in Europe in five categories.

3.5.1 German Analyses

Reports evaluated in this category have a document identification of “EG” and include analyses concerning fuel load changes, advanced reactors, Chernobyl effects, and emergency planning.

3.5.1.1 Fuel Evaluation

The EG-1 document uses the COSYMA code in a recent analysis to evaluate the consequence of varying the amount of enriched fuel in the German BER II reactor. Results are presented in conventional format but show the higher enrichment to have greater consequences. A bar graph display is used to show the isotope group contribution to pathway dose (see Figure B33).

3.5.1.2 Advanced Reactors

The German analysts at KfK have been evaluating the consequences associated with advanced reactor accidents. One such analysis is presented in the EG-2 document

The EG-2 analysis using the COSYMA code represents a major effort and reaches a conclusion that may be somewhat surprising, given the attention to new advanced designs. The report concludes that even with a double containment but without filtration, margins are not large enough to preclude having to take offsite protective actions to meet established dose criteria. This conclusion is reversed if the containment annulus filters are assumed to be effective. Concerning food contamination, even with filters a 2-year ban on some foods for less than a 4 km area is required in 1% of the cases. Dairy products require a larger area ban for 3 months. Figures B34 and B35 provide examples in dose versus time and dose versus distance formats, respectively. Surprisingly, V sequences (with bypass) were not considered in this analysis.

3.5.1.3 Chernobyl Doses

The COSYMA code was used in the EG-3 document to compute doses to German citizens in the USSR after the Chernobyl accident. While the source term and meteorological data are not well known, results of the consequence analysis were in good agreement with the measured radioactive material deposited on the ground. The table in Figure B36 shows this comparison.

3.5.1.4 Filtered Venting

Studies of the effectiveness of filtered containment vents after core melt are presented in the EG-3 document. This paper presented results of more recent calculations using the newer codes. The German Risk Study – Phase B completed in 1989 includes source terms from this venting procedure.

The EG-3 report discussed filtered venting of a large PWR containment prior to reaching the limiting pressure 1 to 2 days after the accident start. The release height was a 120 meter stack. The venting period was 48 hours during which about half of the containment inventory is released. The release was separated into 24, 1-hour puffs separated by 1 hour. The variable trajectory UFOMOD code was used to calculate dispersion to account for wind direction changes.

Several calculations with varying filtration assumptions were made. Use of only an aerosol filter resulted in about a factor of 2 reduction. However, addition of an elemental iodine filtration system resulted in a factor of 50 reduction. Further reduction by adding organic filtration did not result in appreciable benefit. An asymptote in the dose was reached at higher filtration efficiencies due to the noble gas contributions. Milk would have to be banned in large areas (thousands of square kilometers) without iodine filtration. It is concluded that iodine filtration with 99% efficiency and 99.9% for particulates is required for optimal performance.

3.5.1.5 Emergency Planning

The EG-4 and EG-6 documents are in German and were not translated, but they deal with evacuation and countermeasure effectiveness. Display of results are presented that may be unique (see Figures B37 through B39).

3.5.2 Dutch Reports

Three recent evaluations by the Dutch were found that fall into the "EH" document category. All were prepared by KEMA for the Dutch government. Unfortunately, two were written in the Dutch language but are included here because they appear to represent very comprehensive reports that, in part, rely on consequence evaluations.

3.5.2.1 New Reactor-Siting

In 1990, the Dutch prepared a consequence assessment for an imaginary 1,000 Mwe BWR or PWR (see EH-1 document). The study took advantage of the variable trajectory plume segment model in COSYMA and used 2 years of site data. The effectiveness of countermeasures was explored.

Since the study was made just as COSYMA was being developed, some parts of the analysis used UFOMOD German default data, which are different than for Holland. Also, they recommended addition of aquatic models for contamination of lake surfaces. There apparently were also problems with use of long-term intervention doses as a function of pathway. These problems were probably addressed and corrected in more recent versions of COSYMA.

3.5.2.2 Emergency Planning Assumptions

In a number of the documents reviewed during this study, evacuation, intervention, interdiction, shelter and other countermeasure actions are assumed to be taken at various dose levels in the codes. These intervention criteria differ from one country to another and from code to code. In the EH-1 document, there is considerable discussion of how to achieve Dutch criteria within the framework of the early COSYMA (and UFOMOD) codes. This document explains the adjustments that had to be made.

Thus, the users are attempting to use existing codes to emulate the effects of an emergency plan, if properly implemented during an emergency. Many of the European consequence calculations present results in CCDF format of the number of persons evacuated, the amount of food banned and the interdiction area. In the U.S. studies, there is not as much emphasis on these quantities, rather health effects are of more interest.

The Dutch separate the evacuation into groups with the first starting before cloud passage. This was impractical with the COSYMA code that evacuated a keyhole area independent of expected doses. Another factor that had to be considered was that the dose criteria had a range of 50 to 250 mSv. The higher value was selected for the entire area.

Another problem existed concerning integration time (a ground shine issue) for relocation. Only one time was available in COSYMA, yet the Dutch had both 1-year and 50-year intervention levels. They chose to use a 50-year integration time with the higher dose criterion for resettlement divided by 2. The authors indicated that the code was not written for this kind of use.

3.5.2.3 Environmental Effects

Two large Dutch language reports were obtained from KEMA (EH-2 and EH-3) that compute the environmental impact of the each of the two Dutch reactors. These are very well prepared documents that integrate displays in with the text and probably are written for public use (as opposed to highly technical presentations). Only a few types of graphs in CCDF and dose versus distance format are presented in the report as exemplified by Figures B40 and B41. A very small appendix on COSYMA is presented and the conventional data from a consequence study is not found in the report. Since it was not translated, it is difficult to tell if references are made to other separate consequence analysis reports. Individual risk seems to be the subject of more of the plots presented.

3.5.3 Site Selection in England

Several generic English documents were found that were classified in the “EC” category.

The EE-1 document provides a summary of a very large effort conducted in the U.K. in the mid 1980s. They utilized the MARC code to evaluate 23 real and fictitious sites and ranked them from best to worst. An urban center site was also evaluated for comparison. It was interesting to note how the ranking, presented as a series of bar graphics (see Figures B42 and B43) changed for different release categories. The authors noted that the economic model could not yet be used at the time of the study. In general, the U.K. documents rely more on tabular presentation of results.

In 1985, a consequence workshop (EE-2 document) took place with many excellent papers presented. Many were from English authors and the proceedings are referenced here for completeness. One particular display was of interest in which the author plotted cancer fatalities versus time of relocation (see Figure B44).

In 1985, the MARC code was used to study seasonal effects on agricultural consequences (document EE-3). The EE-4 document describes verification and validation of the NRPB models used to track radionuclides through the environment, and the EE-5 document describes the COCO cost model used in CONDOR and COSYMA.

3.5.4 Sizewell and Hinkley Evaluations

In the 1980s, the U.K. experienced a long and complex public inquiry on the safety of operating PWRs at the Sizewell and Hinkley sites. Ultimately, both sites were approved. Based on discussions with technical witnesses involved in the inquiry, the consequence results were well received in the process. Since NUPEC had particular interest in these analyses, nine representative documents for Hinkley and Sizewell were obtained from NRPB and are evaluated below in the "EI" document category. Other documents were requested but were not available.

The English reports tend to present tables showing the amount of food interdicted. This type of information is not found in consequence assessments using MACCS.

3.5.4.1 Degraded Core Analyses for Hinkley

The EI-1 document, prepared in 1987, is very similar to those discussed below. This Hinkley study used the MARC code since MARC-1 was not yet available. The EI-2 through EI-5 documents discussed below, all use the MARC-1 code. The twelve source terms identified in the study as UK1-UK12 were received from Westinghouse based on Levels 1 and 2 analyses.

A rather unusual situation exists for the weather data used in the Sizewell and Hinkley analyses. It was taken from locations several tens of kilometers away from each site. Some studies were done to show this was acceptable. This is unusual for site-specific studies of this type.

The typical pathways and countermeasures were used and results presented in a series of tables found in most U.K. reports in this group. They are complex tables giving considerable information as shown in Figures B45 through B52. A number of CCDFs are also used in the results presentation, including interdicted area, number of people evacuated (i.e., dose above a certain level) and the amount of food restricted (see Figures B53 and B54).

The degraded core assessment was rerun, using MARC –1 as soon as it was available (see the EI-2 document). Results were within about a factor of 2 of MARC results and were explained by changes in dose factors for key isotopes.

3.5.4.2 DBA Accidents Analysis

In the EI-3 document, MARC-1 is used for the Design Basis Accident. Results documented that there was no need for countermeasures other than food bans. Only three source terms were used and many runs were made with no countermeasures in effect. Partial sheltering was used in these studies. The U.K. standard tabular format was used to present results. A CCDF including hereditary effects was also presented (see Figure B55).

3.5.4.3 Containment Bypass Analysis

One of the most evaluated accidents for PWRs is the release through pipes that connect the pressurized system to components outside containment. Eight release categories were identified and evaluated using MARC-1 as reported in the EI-4 document. Surprisingly, degradation of the core was not assumed. Results showed that evacuation was required for some release categories. Output was presented as the standard set including tables and CCDFs.

3.5.4.4 Agricultural Consequences Analysis

In the EI-5 document, an assessment of the effect of crop interdiction was made. The British use criteria for crop bans that are different than for the EC. Three release categories from the degraded and design basis source terms were used. Results showed the lower EC criteria led to lower doses and greater affected areas. It would be difficult to use MACCS for this type of study where interdicted areas and food amounts are presented.

3.5.4.5 Countermeasure Effectiveness

The EI-6 document presents an interesting analysis of countermeasure effectiveness for Sizewell. Plots are presented showing cancer reduction based on time of relocation and return.

3.5.4.6 Early Degraded Core Analyses for Sizewell

A series of degraded core consequence assessment were made in the 1982-1983 time frame for Sizewell. The EI-7 document presents the results of an even earlier analysis with updated source term frequencies. The EI-8 study computes consequences in the very large London population center. A sensitivity analysis is presented in the EI-9 document. Conditional CCDFs were used to characterize consequences in these reports.

3.5.5 Other European Studies

In discussions with code users and during the IAEA Technical Meeting, additional examples of uses of consequence methodology were found. The following presents brief discussions on documents in the “EO” category for these less significant applications.

3.5.5.1 Procedures for Level 3

The EO-1 document is presented first since it was prepared by the European group that is performing consequence assessments. It represents an orderly set of recommendations for conducting a Level 3 analysis and gives guidance for presentation of results.

3.5.5.2 Greek Research Reactor

In EO-2, a Greek organization presents results of a consequence analysis for a research reactor. Of importance here is the comparison of results between PC-COSYMA and MACCS which were both run. An inconsistency in the calculation of total health effects was discussed.

3.5.5.3 VVER Analysis

The Hungarians have used MACCS in the EO-3 document to compute consequences for their Russian VVER plants at the Paks site. This seems to be an incomplete analysis at this point that is somewhat generic (no Level 2 results are apparently available for use as source terms). Ingestion pathway was not used. Some atypical plots of fraction of people whose dose exceeds certain values versus distance are presented (Figure B56). They plan to modify the code to use monthly varying crop data.

3.5.5.4 Czech Republic Activity

The Czech Republic apparently is not using a probabilistic code – rather in EO-4 and EO-6, they report results using HERALD which is not a true Level 3 code capable of providing probabilistic results. In EO-6, the sensitivity to long-term weathering is discussed as a problem with their code.

3.5.5.5 Consequence Calculations in Slovenia

Several analyses are presented in EO-5 for the Krska plant and for Chernobyl. So far the CRAC2 and PC-COSYMA codes have been used. These calculations have confirmed the effectiveness of countermeasures.

3.5.5.6 Norwegian Studies

The Norwegians have conducted many good studies related to long-term effects of contaminated areas for many years. The activity has intensified since Chernobyl. The EO-7 document presents data on an important contamination parameter under winter conditions. This type of information may be of interest in certain areas of Japan.

3.5.5.7 Hong Kong

Although not in Europe, a consequence assessment has been performed for the Daya Bay Nuclear Plant in China. A new code called RADIS was developed for this purpose by a professor at the University of Hong Kong.[12] The code is not probabilistic and was developed primarily to compute doses in population centers using a realistic dispersion model.

3.6 Display of Consequence Results

During the review of the documents collected, one of the major objectives was to evaluate the display of results. To assist the reader, examples were excerpted from the voluminous set of documents and included in Appendix B. An index is provided in the front of the appendix summarizing the contents. The parent document identifier is also given in the Appendix B index.

By far the most common parameter presented is the mean (or expected) number of health effects. This is followed by tables of frequencies for different effects which are often plotted as CCDFs. Examples of many other formats are included in Appendix B.

After searching through more than 100 documents in this study, it appears that Level 3 presentations in the full-scope PRAs prepared by PLG are the most comprehensive (see the UO-2 and UO-3 documents for examples). For most applications in Europe, the most popular displays are in tabular form and/or in the basic CCDF format. PC-COSYMA certainly is more convenient to use and has a very nice color graphics presentation of results.

3.7 Current Status of Consequence Codes

Based on discussions with the code developers and users, a consensus on the current status of consequence codes with emphasis on Europe and U.S. is presented in this section. It should be noted that the OSCAAR code, which is a full-featured code and superior in several respects to other consequence codes, was not included in the scope of this study.

3.7.1 Review of Documents — Code Descriptions

The last two categories in Table 3-2 include documents related to consequence codes reviewed in this study. Those in the "MC" category pertain to the MACCS code and the "OC" category refers to all other codes. Discussions related to these documents fall into several general categories that follow. Code descriptions are included in documents OC-12 and OC-16 for MARC, in documents OC-1 and OC-11 for CONDOR and in documents MC-1 and MC-3 for MACCS. Because of the popularity of COSYMA, many documents were found and are included in the document list identified as OC-4, OC-5, OC-7, OC-8, OC-9, and OC-10. A very good description of PC-COSYMA is provided in the OC-6 document.

3.7.2 Development

As discussed earlier, the major codes in use in the U.S. and Europe are MACCS (and MACCS2), COSYMA and CONDOR. Very little new development has occurred since 1994.

Representatives from the CEC report that COSYMA and PC-COSYMA development has stopped and most CEC funding is directed toward the RODOS real-time code. AEA representatives report that CONDOR is essentially complete and in use primarily in the UK. Some modifications are considered based on User Group feedback.

As regards MACCS2, the major development items (from MC-7) are:

- Expansion of the number of nuclides to include 800 with decay chains of up to six members
- Incorporation of a new preprocessor is used to compute internal dose commitment factors per ICRP-30.
- Addition of a new preprocessor for food ingestion. Food bans are separate for milk and non-milk.
- The evacuation model now allows curved evacuation paths, varying travel speed, and up to three population groups.

Extensive uncertainty and sensitivity analyses have been completed for the U.S. codes (see the MC-2, MC-8 and MC-9 documents), for UFOMOD (used in COSYMA) in documents OC-13 and OC-14, and for MARC (also used in COSYMA) in the OC-15 document.

3.7.3 International Benchmark Exercises

Two international benchmark (or code comparison) exercises have been completed, one in the early 1980s[6] and the latest one in the early 1990s.[7] These are considered useful for quality assurance and educational purposes. Generally, the test problems do not exercise many features of the better codes (e.g., curved plumes, variable evacuation, and complex food chains). Several reports include results from the second exercise (see documents MC-5, OC-2 and OC-3).

In general, the codes compared fairly well in most categories except for MACCS, which had a very low mean population dose due to ingestion. This was explained by two MACCS characteristics related to uniform food production and crop ban criteria.

MACCS was not able to use the prescribed benchmark food distribution data defined in considerable spatial detail.

3.7.4 Code Comparison

Table 3-3 provides a detailed comparison of the code capabilities including CONDOR, MACCS2 and PC-COSYMA, as well as comments related to CRACIT (CRACEZ) features.

3.7.5 User's Groups

Active user groups have been established for MACCS, CONDOR, and COSYMA. These groups have held meetings to discuss problems, perform benchmark calculations, and identify development needs.

Twelve countries were present at the last COSYMA meeting reported in the OC-7 document prepared by KEMA. This group has a newsletter and holds training sessions.

The CONDOR users group meets regularly and operates a help "hot line." The group members are mostly from the U.K. and Holland. Recommendations for future use are presented.

The MACCS group is referred to as an "international" group and includes primarily those countries that participated in the second benchmark exercise.

Though not a formal user's group, the IAEA sponsors technical meetings concerning Level 3 methods and applications. This group has prepared a guide on procedures for preparing Level 3 analyses for nuclear plants (see the EO-1 document).

3.7.6 User Satisfaction/Problems

In discussions with the code users questions were asked concerning problems. Also, in reviews of the reports and meeting papers, an attempt was made to determine if problems had occurred. Actually, very few indications of problems were found during the evaluation.

Probably the most comprehensive comments found were those by a MACCS user in Scandinavia (see document MC-4 and MC-6). These issues were presented at a recent international MACCS users group meeting.

Table 3-3
Comparison of Consequence Codes

Mode/Data	COSYMA (PC-COSYMA)[1]	CONDOR	MACCS (MACCS2)[2]	Comments Concerning Significant Features of CRACIT and CRACEZ
Developer	CEC (KfK and NRPB)	SRD (UK), Nuclear Electric (UK), and NRPB (UK)	U.S. Nuclear Regulatory Commission and DOE, U.S.	PLG
Atmospheric Dispersion Model(s)	Segmented Gaussian plume model (MUSEMET) with time-variant meteorology. Beyond 60 km trajectory puff model (MESOS) with time and space-variant meteorology. Note: PC-COSYMA has only MUSEMET plume segment, variable trajectory model.	Multiple "straight line" Gaussian plume with time-variant meteorology.	Gaussian straight-line. Plume width increases with time.	CRACIT: Variable trajectory, plume segment hourly changes. CRACEZ: Particle-in-cell 3-D terrain dispersion with 15-minute changes.
Plume Rise Model	Briggs out to 60 km then none.	Briggs	Briggs	Briggs
Dry Deposition Model	Source depletion.	Source depletion.	Modified Chamberlain source depletion method: particle size distribution with specific dry deposition velocity for each size.	CRACIT: Source depletion two step in each spatial interval. CRACEZ: Based on particle density in lower cells.
Wet Deposition Model(s)	Rain intensity dependent wet deposition (power law with exponents): rain is spread over the plume segment. At large distances, rain is spread over trajectory according to the data from local weather conditions.	Washout model with exponential depletion or "effective lambda model" which spreads wet deposition over the length of the plume.	Washout following Brenk and Vogts. Exponential source decay; washout onto all spatial elements located under the cloud segment.	CRACIT: Washout using power law for four rain rates. CRACEZ: Full column removal using power law.
Weather Data Required	Hourly data at one location within 60 km. 3-hourly weather reports from all stations beyond 60 km. (Apparently not used in PC-COSYMA).	Hourly data at one location.	Hourly data: wind speed, wind direction, atmospheric stability, precipitation rate; seasonal data; elevation of the mixing layer for fall, winter, spring, summer at one location.	CRACIT: Hourly data speed, direction, stability, precipitation rate and lid height up to 20 towers. CRACEZ: Same as above only hourly or 15 minutes.

[1]Notes are provided where PC-COSYMA differs from COSYMA.
[2]Notes are provided where MACCS2 differs from MACCS.

Mode/Data	COSYMA (PC-COSYMA)[1]	CONDOR	MACCS (MACCS2)[2]	Comments Concerning Significant Features of CRACIT and CRACEZ
Sampling Scheme	Stratified sampling with data grouping user defined. Other possible schemes: user-specified starting time, cyclic sampling, user-specified constant weather. At large distances, starting times cyclic every 61 hours, equal probability. Other possible schemes user specified starting time, cyclic sampling. Note: In PC-COSYMA there are fewer sampling options.	Stratified sampling with data grouping user defined.	120 hours of data represent a "weather sequence"; 5 options: constant weather, user-specified weather sequence (120 hours), user-specified start stratified random sampling, and structured Monte-Carlo sampling using sequence binning based on initial conditions and precipitation characteristics; "boundary weather" defined for the plume segments remaining within the computational map boundary after 120 hours.	Codes use stratified sampling combined with "tails search" to identify high consequence scenarios.
Source Term	Up to nine release starts per run. Note: PC-COSYMA allows up to three.	Multiple straight-line releases.	Up to four straightline releases. Note: MACCS2 has expanded nuclide list.	CRACIT: Up to four releases. CRACEZ: Up to 96 15-minute releases.
Cloud Gamma Model	Semi-infinite with precalculated near range correction factors belonging to the set of sigma values.	Semi-infinite or semi-infinite with geometrical correction factors and data from ICRP5l, or direct dose look-up table computed using full 3-D integration of Gaussian plumes.	Semi-infinite cloud approximation model with a finite cloud correction is used during the emergency phase (up to 1 week). The dose conversion factors file is based on data provided by the Oak Ridge National Laboratory, U.S.; the file contains conversion factors for 19 organs and 60 radionuclides.	CRACIT: Same as MACCS. CRACEZ: Plume is separated into 3-D cylindrical volumes that "shine" independently on receptor. Six energy groups are used.
Ground Gamma Model	Doses from material in soil, allowing for penetration into soil and shielding by overlying soil, penetration is described using a four compartment model.	NRPB's soil compartment model, which accounts for migration in and attenuation by the soil, and data from ICRP5l.	Emergency phase: time integration of a linear ramp during plume passage and an exponential decay function afterwards; lateral concentration distribution is taken into consideration. Intermediate and long-term phases: in addition to decay, Gale's weathering function is used.	Contributions from deposits are accumulated on polar fine grid.

[1]Notes are provided where PC-COSYMA differs from COSYMA.
[2]Notes are provided where MACCS2 differs from MACCS.

Mode/Data	COSYMA (PC-COSYMA)[1]	CONDOR	MACCS (MACCS2)[2]	Comments Concerning Significant Features of CRACIT and CRACEZ
Inhalation Dose Model	Dose conversion factor based on NRPB's PEDAL model. Data are compatible with ICRP30/ICRP48 and ICRP56 in part.	Dose conversion factors based on NRPB's PEDAL model, data are compatible with ICRP30 and ICRP48.	Immersion (cloud) inhalation. Emergency phase; acute and lifetime doses; radionuclide lung clearance classes suggested by ICRP30 are used.	Dose factor tables are user defined.
Resuspension Model	Three-term exponential decay/ weathering factor.	Garland's resuspension factor model.	a) emergency phase: one-term exponential decay resuspension factor and a single half-life are used; calculated only after plume passage. b) long-term phase: three-term exponential decay/weathering resuspension factor is used.	--
Ingestion Dose Model	Dose conversion factors based on NRPB's PEDAL model data are compatible with ICRP30/ICRP48 and ICRP56 in part.	Dose conversion factors based on NRPB's PEDAL model, data are compatible with ICRP30 and ICRP48.	Societal dose is calculated based on the total amount of radioactive material ultimately consumed by the population; six radionuclides are used: Sr-89, Sr-90, Cs-134, Cs-137, I-131, and I-133. Note: MACCS expanded number of isotopes and decay chains.	--
Food Chain Model	Uses data, from dynamic compartment- type models FARMLAND (NRPB) and ECOSYS (CSF); up to 15 terrestrial food stuffs; seasonal effects are accounted for by using food chain data for a deposit in January to represent winter conditions and in July to represent summer conditions.	NRPB's FARMLAND model with consequences predicted assuming that the accident occurs in June or Nuclear Electric's FOODWEB model based on four seasons of the year.	Ingestion of contaminated food and water is modeled; up to 10 food categories and 10 nuclides (without daughter isotopes) can be used; a concept of an annual "food basket" for a maximally exposed individual is used in calculations of countermeasures; direct deposition during growing season and root uptake are calculated; animal and direct crop consumption chains are considered. Note: MACCS2 uses COMIDA food chain model.	CRACEZ: Design for early phase and has no long-term model.

[1]Notes are provided where PC-COSYMA differs from COSYMA.
[2]Notes are provided where MACCS2 differs from MACCS.

Mode/Data	COSYMA (PC-COSYMA)[1]	CONDOR	MACCS (MACCS2)[2]	Comments Concerning Significant Features of CRACIT and CRACEZ
Short-Term Countermeasures	Evacuation, sheltering, and decontamination of individuals within fixed areas or based on dose limits; distribution of stable iodine within fixed area. Note: PC-COSYMA has only one population group for evacuation and sheltering.	Evacuation, sheltering, distribution of stable iodine and decontamination of individuals within fixed areas or based on dose limits or concentration levels (not sheltering).	Evacuation and sheltering based on time and distance from the plant; two temporary relation cohorts based on doses, times after plume arrival, and distance from the plant (only for the region beyond the sheltering zone). Note: MACCS2 includes nonradial evacuation paths and variable speed with time. Up to three cohort groups follow separate paths.	CRACIT: Variable trajectory evacuation (time and space) single cohort. CRACEZ: Multiple cohort variable time and space using link-node highway network model with 1-minute updates.
Long-Term Countermeasures	Relocation based on dose limits, land decontamination and sheltering only within fixed area. Criteria for initiation and withdrawal of relocation can be different.	Relocation based on dose limits and land decontamination.	Relation based on dose limits; land and property temporary interdiction or condemnation; three levels of decontamination. Intermediate phase (up to 1 year after emergency phase); only temporary relocation of people based on total dose through ground shine and resuspension inhalation.	--
Food Countermeasures	Food bans based on concentration levels in food or dose levels. Criteria for initiation and withdrawal of food bans can be different.	Food bans based on concentration levels in food or dose levels.	Food bans, farmland decontamination, temporary interdiction, or condemnation based on the ingestion dose limits for a maximally exposed individual. Note: MACCS2 mitigative actions are triggered separately for milk and non-milk crop.	--
Early Effects	Hazard functions based dose-rate dependent values for lung and bone marrow.	Hazard functions based on recommendations of NRPB (NRPB-R226).	Two-parameters Weibull hazard function with threshold (Evans, J. S., NUREG/CR-4214, 1986 and 1989); quality factors for different radiation types are imbedded into dose conversion factors.	--

[1]Notes are provided where PC-COSYMA differs from COSYMA.
[2]Notes are provided where MACCS2 differs from MACCS.

Mode/Data	COSYMA (PC-COSYMA)[1]	CONDOR	MACCS (MACCS2)[2]	Comments Concerning Significant Features of CRACIT and CRACEZ
Late Effects	Linear dose risk functions based on recommendations and data from GSF. Note: PC-COSYMA has reduced data set to limit file space.	Linear function with a dose and dose-rate effectiveness factor (DDREF) at low doses and low dose rate. Based on recommendations of NRPB (NRPB-R226).	Piecewise linear function dose reduction factors are used for low dose or low rate exposures (total dose less than 0.2 Gy or dose rate lower than 0.1 Gy/hr); no dose threshold for latent cancer fatalities. (NUREG/CR-4214, Rev. 1, Part II, Addendum 1).	--
Economic Cost Model	NRPB's COCO-l model that evaluates the cost of countermeasures (evacuation, relocation, decontamination, sheltering, food bans) and also the cost of health effects in the exposed population.	NRPB's COCO-l model that evaluates the cost of countermeasures (evacuation, relocation, decontamination, sheltering, food bans) and also the cost of health effects in the exposed population.	Early phase: costs of evacuation and relocation intermediate phase: cost of relocation; long-term phase; costs of decontamination, temporary/permanent interdiction of farmland and urban property, crop/milk disposal. Note: MACCS2 includes cost interdicted food.	--
Output	Tabular Note: PC-COSYMA has graphics display of pie charts and polar grid to illustrate spatial effects and contributors to results.	Tabular	Tabular	Suite of post-processors that produce graphics output CCDF, risk versus distance, dose versus distance, risk maps.

[1]Notes are provided where PC-COSYMA differs from COSYMA.
[2]Notes are provided where MACCS2 differs from MACCS.

Others indicated that COSYMA was complex and may not be appropriate for less rigorous analyses. PC-COSYMA seemed to be easy to run.

CONDOR apparently has a nice user interface using "key words" and many defaults and is not as hard to setup and run compared with COSYMA.

Results of a COSYMA user's questionnaire were reported in the OC-5 document, indicating that the users had relatively few significant problems. It was suggested that the evacuation timing data needed to be simplified.

3.7.1 Code Distribution

The extent of use of each code was reviewed during discussions with both U.S. and European representatives. Official versions of MACCS are available through the Energy, Science and Technology Software Center (ESTSC). Prior to the establishment of this code center several years ago, copies were available through the National Energy Software Center (NESC). It would be appropriate to assume that the current active users would be using newer versions received from the ESTSC. More than 30 countries are reported to have received copies of COSYMA (or PC-COSYMA).

3.8 Comments on Liquid Pathways

Most consequence analyses concentrate on the airborne and terrestrial pathways, plume shine, ground shine from deposition, inhalation and ingestion. However, several liquid consequence analyses have been performed and are briefly discussed below.

In 1982, the Indian Point PSA reported the results of an analysis where the core was assumed to melt through the bottom of containment and fission products entered the groundwater. After many hundreds or thousands of years, they eventually entered the river and pathways to man. It was concluded that ample time was available to interdict the source and prevent the long-term entry into the environment.

Researchers in Norway have determined that doses through the aquatic pathways (e.g., fish) from deposition of airborne material from Chernobyl are significant. This is due to the ultimate deposit of radionuclides in sediment where fish have access. Apparently, doses due to drinking water are not as significant as those due to fish ingestion by a large factor.

It was not determined to what extent codes had adequate treatment of liquid pathways although CONDOR and COSYMA apparently have some capability in this regard and there are plans for MACCS2.

3.9 Conclusions

The most intense development and application of consequence code methodology has been in Europe primarily by U.K. and German groups. In the last 5 years, they have combined their efforts to produce the COSYMA code with CEC funding. Thus, while CONDOR is being actively used in the U.K., most other European Community countries are using COSYMA (or the more user-friendly PC-COSYMA). There is also significant use of the MACCS code in several European countries.

Only limited active development and/or improvement of consequence codes was found in any country. The CEC is primarily supporting development of real-time consequence codes such as RODOS. No further significant funding for COSYMA or PC-COSYMA is planned.

Users report difficulties in data management for the more complex foodchain and economic models for their applications in some cases. Generally, COSYMA is considered to be very complex and not particularly easy to use (and is long running). PC-COSYMA is much easier to use and does not compromise models that are of importance to most consequence assessments. The European codes tend to use detailed default data that is only applicable in Europe.

The most sophisticated plume trajectory models are in COSYMA, CRACIT and CRACEZ. The most sophisticated long range transport model is in COSYMA. The better foodchain models are in CONDOR and COSYMA. The most comprehensive displays of results are found in PC-COSYMA and CRACIT (and CRACEZ).

Each of these codes has specific application targets. All are generally applicable, having been benchmarked in international trials. There is no code that does all things.

4

CONCLUSIONS

Based on the results of this state-of-the-art assessment of the prior use of Level 3 analyses over the last 25 years, the following conclusions are drawn:

1. Level 3 analyses use base technology that has been benchmarked with a broad range of experiments, thus validating the technical methods and approaches used to determine public health and safety impacts resulting from the operation of nuclear power plants.

2. Level 3 analyses must be developed on a site-specific basis to be meaningful insights on those factors that reflect, positively and negatively, on public heath and safety levels.

3. The same basic technology is used widely in a range of industries for evaluating the consequences of real and postulated releases of radiological, biological, and hazardous chemicals, and forms the basis for the emergency response planning in case of such emergencies.

4. There are a number of computer codes in the U.S. and worldwide that can be used for developing Level 3 models. All have been benchmarked in international benchmarking exercises. Each code has its particular application specialty.

5. The accuracy of Level 3 calculations is mainly driven by the accuracy of the Level 1 and Level 2 model outputs. For the Level 3 results to be reliable and dependable, the remainder of the PSA must be realistic and validated by comparison to operating experience.

6. In many cases, the overall uncertainties of a PSA are dominated by those introduced in the Level 2 predictions in the form of phenomenological uncertainties in the core melt process and containment behavior. While Level 3 calculations introduce some additional levels of uncertainty, the industry has valid techniques for treating the uncertainties. Other industries using this technology deal effectively with similar uncertainties.

7. The existing Level 3 calculational techniques are certainly adequate for the risk-informed, performance-based pilot projects underway and eventual use in

comparisons to the quantitative health objectives (QHO) as defined in the 1986 NRC policy statement on Safety Goals.

8. The QHOs should provide an alternative basis for the regulation of commercial nuclear power plants in the U.S because:

    a. They are one of the only means we have to assess the impact of operation of such plants on the health and safety of the public.

    b. They are consistent with the health and safety assessments used in other industries in the U.S. and other nuclear plant regulators worldwide.

    c. The QHOs provide an established reference point for determining the acceptable levels of "how safe is safe enough" with respect to public health and safety.

5

RECOMMENDATIONS

Based on the results of this state-of-the-art assessment, the following recommendations are made:

1. The minimum quality and content requirements of the Level 1 and 2 portions of a PSA to qualify the results of a Level 3 consequence analysis must be specified.

2. Expand the NEI Pilot Project for risk-informed, performance-based regulations sufficiently to confirm the usefulness of such an approach as an alternative means for commercial nuclear power plant regulation.

3. Develop a generalized module to facilitate the use of Level 3 analyses in the pursuit of risk-informed, performance-based regulations.