Filed: March 20, 1984

75

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

before the

ATOMIC SAFETY AND LICENSING BOARD

In the Matter of PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE, et al. (Seabrook Station, Units 1 & 2)	Docket Nos. 50-443 OL 50-444 OL

APPLICANTS' RESPONSE TO "NECNP MOTION
TO FILE SUPPLEMENTAL CONTENTION ON
NEW HAMPSHIRE STATE AND LOCAL EMERGENCY RESPONSE PLANS"

NECNP has moved for the late admission of a contention challenging the as-yet undesigned public alert system on the grounds that maybe that system will depend on off-site power, maybe the off-site power will fail from time to time, and therefore the system is inadequate under the Regulations. For the reasons set forth herein, the motion should be denied.

As the Board is aware (and NECNP admits) the design for the public notification system has not yet been

DS03

published. The emergency plans presently contain what amounts to a "blank" for this item. In fact, a design study has recently been completed, and, we understand, will be contained in an amendment to the New Hampshire State Plan to be published in the near future.

The proposed new contention says, in effect, that "_If_ the public alert system depends upon off-site AC power for operation, and _if_ the offsite AC power system may from time to time fail, then the public notification system is inadequate under the applicable regulations." The contention is hopelessly hypothetical, and therefore fails the first teaching of the _Catawba_ case. Moreover, if the contention were permissible in the hypothetical form, it would be hopelessly out of time, since nothing in the Seabrook Station Probabalistic Safety Assessment ("SS-PSA"), which is the sole proffered basis for excusing tardiness, supplies any new information.

The first prong of the contention is that _maybe_ the public notification system, when the design is published, will depend upon offsite AC power for operation. NECNP _Motion_ at 2: "_To the extent that any of these systems depend upon offsite power sources to_

operate . . . ." (Emphasis added.) In fact, NECNP doesn't know whether there is any basis on this score in fact; there is no way that the Board can know whether there is any basis on this score in fact; and the contention is hopelessly premature and hypothetical. NECNP proposes to deal with these problems by relying upon later amendment once the system design is available: "The issues will be clarified, however, when the design of the audible alert system, showing the extent to which the system relies on sirens and other offsite power-dependent systems, is submitted." NECNP Motion at 5. Such a contention is inadmissible (without regard to timeliness), and the notion of admitting a contention now in order to wait and see if it has any basis when later information is available has been explicitly rejected. Duke Power Company (Catawba Nuclear Station, Units 1 and 2), ALAB-687, 16 NRC 460, 466-67 (1982), rev'd on other points, CLI-83-19, 17 NRC 1041 (1983).

The second prong of NECNP's proposed contention is that there is some correlation -- some connection -- between a loss of offsite power to the circuits that feed residential houses and street lighting poles, and

the circuits that provide off-site AC power to Seabrook Station for plant operations. It is on the basis of such an assumed connection that NECNP infers a correlation between the happening of an emergency at Seabrook and the need for the off-site notification system. NECNP Motion at 2-3 & n:

> "The [SS-PSA] demonstrates that over half of the accidents at Seabrook leading to a significant radioactive release (and thus requiring an emergency response) would involve a loss of offsite power. Therefore the sirens and any other notification devices dependent upon offsite power are likely to be disabled and rendered useless in an emergency at Seabrook."

(Emphasis added; footnote omitted.) There is, however, no basis for the assertion contained in NECNP's motion, nor does the SS-PSA supply any.[1] For this additional

---

[1] Indeed, what the SS-PSA does show is that the probability of an accident leading to a significant offsite release is quite rare, and that if it were to happen, the time before the offsite release occurs is so long that it is difficult to imagine that any inefficiencies in notification due to loss of AC power-depended devices would have any effect at all:

> "It is clear from the Seabrook Station Probabilistic Safety Assessment that events leading to fatalities due to exposure to radioactive material following an accident are indeed very rare. An 'upper bound' esimtate on the frequency of events that result in a small number of fatalities, say 1 to 100, is one such every half

reason, the contention lacks basis and is therefore inadmissible, without regard to timeliness.

Finally, even if the basis and specificity rules permitted a contention to be advanced in the form in which this one is offered, this contention would be inadmissible because it is untimely. If it is sufficient to say that some sirens need AC power and sometimes AC power goes out, then this contention could have been admitted months -- maybe even years -- ago. If there is no need to await the public notification system design, then there was no need to wait for the New Hampshire State Plan. The PSA adds nothing to the hypothetical (nor, indeed, does it even support it).

---

million to a million years. The 'best' estimate for the frequency of such an event is one every 30 million to 50 million years. Thus for all practical purposes, there is no appreciable risk of early fatalities from the operation of Seabrook Station. The reason for essentially no early fatality risk is related to the very high strength of the Seabrook Station containment. The ultimate strength was analyzed to be above 170 pounds per square inch (gauge) -- nearly three times the design pressure. Thus, containment failure is almost an impossibility. About the only accidents that directly fail the containment and contribute to risk are those that occur some 2-1/2 days or so following a damaged core. Such accidents must result in a loss of all containment heat removal capability. As is observed below, these accidents

For these reasons, the pending motion should be denied.

Respectfully submitted,

Thomas G. Dignan, Jr.
R. K. Gad III
Ropes & Gray
225 Franklin Street
Boston, Massachusetts 02110
Telephone: 423-6100

Dated: March 20, 1984

---

affect, for the most part, only the delayed health effects."

Seabrook Station-Probabilistic Safety Assessment, Summary Report at 17-18. For the convenience of the Board, a copy of this Summary Report is attached to this Response.

CERTIFICATE OF SERVICE

I, Robert K. Gad III, one of the attorneys for the Applicants herein, hereby certify that on March 20, 1984, I made service of the within APPLICANTS' RESPONSE TO "NECNP MOTION TO FILE SUPPLEMENTAL CONTENTION ON NEW HAMPSHIRE STATE AND LOCAL EMERGENCY RESPONSE PLANS" by mailing copies thereof, postage prepaid, to:

Helen Hoyt, Chairperson
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory
Commission
Washington, DC 20555

Ms. Diana P. Randall
70 Collins Street
Seabrook, NH 03874

Dr. Emmeth A. Luebke
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory
Commission
Washington, DC 20555

William S. Jordan, III, Esquire
Harmon & Weiss
1725 I Street, N.W.
Suite 506
Washington, DC 20006

Dr. Jerry Harbour
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory
Commission
Washington, DC 20555

G. Dana Bisbee, Esquire
Assistant Attorney General
Office of the Attorney General
208 State House Annex
Concord, NH 03301

Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory
Commission
Washington, DC 20555

Robert G. Perlis, Esquire
Office of the Executive Legal
Director
U.S. Nuclear Regulatory
Commission
Washington, DC 20555

Atomic Safety and Licensing
Appeal Board Panel
U.S. Nuclear Regulatory
Commission
Washington, DC 20555

Robert A. Backus, Esquire
116 Lowell Street
P.O. Box 516
Manchester, NH 03105

Philip Ahrens, Esquire
Assistant Attorney General
Department of the Attorney
General
Augusta, ME 04333

Charles Cross, Esquire
Shaines, Madrigan & McEachern
25 Maplewood Avenue
P. O. Box 366
Portsmouth, NH 03842

Ms. Roberta C. Pevear
Assistant Attorney General
the Town of Hampton Falls
Drinkwater Road
Hampton Falls, NH 03844

Mrs. Sandra Gavutis
Assistant Attorney General
the Town of Kensington
RFD 1
East Kingston, NH 03827

Senator Gordon J. Humphrey
U.S. Senate
Washington, DC 20510
(Attn: Tom Burack)

Senator Gordon J. Humphrey
1 Pillsbury Street
Concord, NH 03301
(Attn: Herb Boynton)

Mr. Donald E. Chick
Town Manager
Town of Exeter
10 Front Street
Exeter, NH 03833

Anne Verge, Chairperson
Board of Selectmen
Town Hall
South Hampton, NH 03827

JoAnn Shotwell, Esquire
Assistant Attorney General
Department of the Attorney General
One Ashburton Place, 19th Floor
Boston, MA 02108

Mr. Patrick J. McKeon
Selectmen's Office
10 Central Road
Rye, NH 03870

Mr. Calvin A. Canney
City Manager
City Hall
126 Daniel Street
Portsmouth, NH 03801

Mr. Angie Machiros
Chairman of the
Board of Selectmen
Town of Newbury
Newbury, MA 01950

Mr. Richard E. Sullivan
Mayor
City Hall
Newburyport, MA 01950

Town Manager's Office
Town Hall
Friend Street
Amesbury, MA 01913

Brian P. Cassidy, Esquire
Regional Counsel
Federal Emergency Management
  Agency - Region I
442 POCH
Boston, MA 02109

Brentwood Board of Selectmen
RFD Dalton Road
Brentwood, NH 03833

Gary W. Holmes, Esquire
Hclmes & Ells
47 Winnacunnet Road
Hampton, NH 03841

Robert K. Gad III

SEABROOK STATION
PROBABILISTIC SAFETY ASSESSMENT

SUMMARY REPORT

B. John Garrick
Study Director

Prepared for
PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE
Manchester, New Hampshire
and
YANKEE ATOMIC ELECTRIC COMPANY
Framingham, Massachusetts
December 1983

Pickard, Lowe and Garrick, Inc.
Engineers • Scientists • Management Consultants
Irvine, CA Washington, DC

TABLE OF CONTENTS

LIST OF TABLES AND FIGURES

INTRODUCTION

A full scope probabilistic risk assessment has been performed for the Seabrook Station nuclear power plant. The study provides an independent assessment of the health and safety risk to the public based on unique features of the plant including its location, design, plans for operations, maintenance, and emergency response. The methods and results of the assessment are documented in detail in the Main Report and its Appendices. The purpose of this Summary Report is to describe with a minimum of technical terms the methods of probabilistic risk assessment and their application to Seabrook Station, and to present a brief summary of the results of the study.

HISTORY OF PROBABILISTIC RISK ASSESSMENT

Probabilistic risk assessment (PRA) is currently the most systematic and comprehensive way to determine the risk of complex technical systems. PRA's unique value stems from its methodology that integrates many technical disciplines and analysis techniques to provide a quantitative assessment of risk. PRA encompasses the engineering disciplines, the mathematics of probability and statistics, human factors analysis, computer models, mathematical models, and extensive data bases.

The safety analysis discipline of PRA evolved from earlier efforts to enhance the effectiveness of engineered systems. For example, following the First World War, multiengine aircraft were introduced as a means of improving the reliability of aircraft. The early 1940s saw the introduction of mathematical models for use in reliability engineering. Use of such models enabled General Motors to extend the useful life of their traction motors used in locomotives from 250,000 to 1,000,000 miles. During the 1950s, new industries with increasingly complex systems stimulated the need for more sophisticated reliability analysis.

Quantitative failure analysis and reliability modeling were developed further in support of the manned space flight and defense programs of the 1960s. The backbone of modern PRA methodology is event tree and fault tree analysis which was developed at such places as Bell Laboratories, the National Aeronautics and Space Administration (NASA), and the Department of Defense. Event trees are used to identify and examine accident scenarios, while fault trees are the basis for calculating the probabilities of those sequences.

In recent years, PRA methods have developed rapidly and are being applied with increasing frequency to complex industrial systems. Examples of these include commercial aircraft, chemical plants, NASA programs, and nuclear power plants.

RISK ASSESSMENT STUDIES OF NUCLEAR POWER PLANTS

PRA methodology has been extensively applied to nuclear power plants. The first full scope PRA performed for commercial nuclear power plants was the Reactor Safety Study (RSS). This study was commissioned by the Nuclear Regulatory Commission (NRC) and directed by Dr. Norman C. Rasmussen of the Massachusetts Institute of Technology. The study was completed in 1975 after 3 years of work; its charter was to analyze the generic risk from nuclear power plants. Consideration of plant specific factors was limited to two plants.

THE REACTOR SAFETY STUDY

The benchmark Reactor Safety Study report, called WASH-1400, was thoroughly reviewed and critiqued in the years following its publication. In response to some of these critiques, the NRC established an independent review group to evaluate the critiques and challenges. This independent review group was chaired by Dr. Harold W. Lewis, Professor of Physics at the University of California, Santa Barbara. The review group's findings, published in a report popularly referred to as the Lewis Committee Report, exemplify the scientific review process. While the Lewis Committee expressed some criticism of the RSS, it concluded that the study was competently performed in good faith and employed sound methodology.

> We find that WASH-1400 was a conscientious and honest effort to apply the methods of fault tree/event tree analysis to an extremely complex system, a nuclear reactor, in order to determine the overall probability and consequences of an accident.
>
> We do find that the methodology, which was an important advance over earlier methodologies applied to reactor risks, is sound....

This critical review of WASH-1400 also identified weaknesses which included inadequate treatment of common cause failures (simultaneous failures of more than one system) and understatement of some of the uncertainties. More recent probabilistic safety studies in this country and in Europe have built on the foundation of the Reactor Safety Study and have incorporated advances that addressed these critiques as well as other advances.

PLANT SPECIFIC PRAs

PRA is an evolving scientific discipline which continues to expand and formalize new insights into risk management. Two recent plant specific studies performed for the Zion and Indian Point nuclear plants exemplify the state of the art at the time the Seabrook Station PRA got started. PRA advancements incorporated into the Indian Point Study included:

- Use of matrix formulations which make the final process of assembling the information from the different parts of the analysis more visible and therefore easier to understand. This method of risk information

assembly also reveals more visibly the factors that contribute to risk and therefore makes the information more useful in risk management.

- Development of a master logic diagram to assist in identifying initiating events.

- Introduction of a framework in which uncertainty is included as an integral part of the presentation of risk.

- An expanded data base, including plant and site specific data and inproved metnods for quantifying uncertainties due to lack of data.

- More accurate risk models for earthquakes, fires, and winds.

- Improved methods for analysis of damaged core phenomena and the role of engineered safety systems during an accident.

- Use of mathematical analysis to express the extent of conservatism in the radioactivity release values which were calculated for the RSS methodology.

- Use of a site specific accident consequences model which allowed for changes in radioactive plume trajectory during the course of a release.

RISK ASSESSMENT METHODS

Modern PRA embraces rigorous logic, computer models, reliability theory, systems analysis, human factors analysis, and the mathematics of probability and statistics. These and the scientific and engineering disciplines are integrated into a formal process that addresses the two components of risk: likelihood and consequence. In general terms, risk is defined and quantified by answering the following three questions:

1. What could happen; i.e., what accident sequences or scenarios are possible?

2. What is the likelihood of these scenarios?

3. What would their consequences be?

ACCIDENTS

The first question is essentially a "what if" question: "what if" some piece of equipment fails; "what if" some error is made; "what if" there is an earthquake, a flood, or a fire? The systematic and comprehensive nature of PRA requires a spectrum of technical experts and the application of rigorous logic to answer this question. The major steps are:

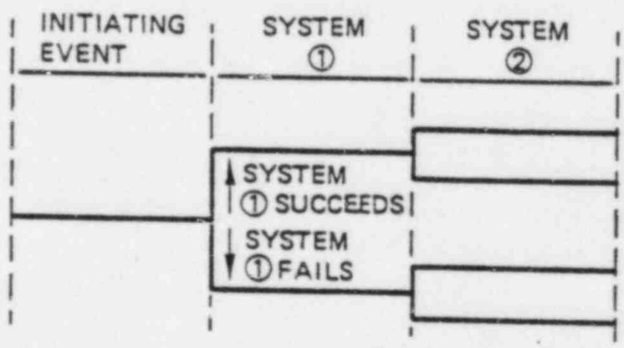
- Develop detailed understanding of the equipment, systems, and structures to be analyzed. In addition, knowledge of procedures,

maintenance, and other aspects must be integrated into a plant model. The detail required includes not only the operation of equipment and systems, but their interrelationships, knowledge of wiring, piping, spatial relationships, and so on.

- Identify "initiating events"; that is, those events that could start an accident scenario. While literally thousands of initiating events are considered, they are discussed as two major classes; "internal" and "external." Internal events are those which originate with malfunctions or failures of plant equipment or systems including those caused by operator error. External initiating events originate from other causes and include earthquakes, fires, high winds, and floods.

- Analyze the possible responses of different systems to initiating events. This process entails development of rigorous logic models which identify all possible combinations of success or failure states for each piece of equipment or system affected. These logic models are described in risk analysis by the use of event trees. Following is a sample event tree:

EVENT TREE CONVENTION:



The event trees for a complex plant trace literally billions of possible scenarios. The purpose of developing trees is to trace sequences leading to situations which pose a threat to human health or to the environment. It should be noted that while it may not be possible to identify absolutely all possible sequences, the approach is to be as complete as possible and then make allowances for scenarios not identified.

LIKELIHOOD

Answering the second question, "What is the likelihood of these accident scenarios?" requires careful analysis of how failures can occur and how likely it is that a given safety system will fail. The likelihood of success or failure of individual systems is investigated using fault tree diagrams, reliability block diagrams, and cause tables. Fault trees and reliability block diagrams express the logical relationships between the

functioning of a system and its subsystems and components and then identifies those factors which could lead to the failure of the entire system. Figure 1 shows an example of a reliability block diagram for the instrument air system and its support systems at Seabrook Station.

The purpose of fault tree analysis is to consider the system in a degree of detail such that statistical evidence can be used to determine the likelihood of failure of components and subsystems which are then used to calculate the likelihood of failure of the entire system. The potential for common cause failures (that is, simultaneous failures of more than one component or system) are also evaluated. In fact, much of the effort that is required to perform such a systems analysis is devoted to the identification and analysis of these "common causes."

The expression "likelihood of failure" embodies two concepts: frequency and uncertainty. Frequency is a measure of how often some event can happen, whereas uncertainty expresses how sure we can be about the frequency estimate. Estimates of the frequency of failures are based on information about the past operation of components, systems, and structures. For each component or system, a range of failure rates (frequencies of failure) is considered and a confidence level for each failure rate is determined. Quantification and representation of uncertainty is central to probabilistic risk assessment. The importance of displaying uncertainty lies in the ability to express the state of knowledge about the system being analyzed in quantitative terms. Quantifying the state of knowledge in each step of the risk assessment process facilitates logical and consistent analysis of both frequent and rare events. The amount of data available affects the confidence that can be expressed in the frequency estimates. If there is a large amount of data available about a given system, a high degree of confidence can be associated with the frequency estimates. Conversely, if historical data are sparse or sketchy or not fully relevant, the level of confidence in the estimates will be low; i.e., more uncertain.

CONSEQUENCES

The next major task is to determine what the consequences of given accidents would be. Assessment of consequences is not a straight, deterministic process. Even for an accident of extreme severity, there is a range of potential consequences. For example, in a severe aircraft accident, a crash, the consequences in terms of loss of life and injury can vary. In some cases, there might be no fatalities or major injuries, but in other similar crashes, the consequences could be extreme. In other words, there is some degree of uncertainty associated with the consequences. This uncertainty is analyzed in a manner similar to the uncertainty associated with frequencies of failures. For an accident of a given severity, a range of damage levels is considered and, for each damage level, a probability or confidence level is calculated. The confidence levels associated with the frequency of accident estimates and the confidence levels associated with the potential consequences are incorporated into the final statement of the risk assessment.

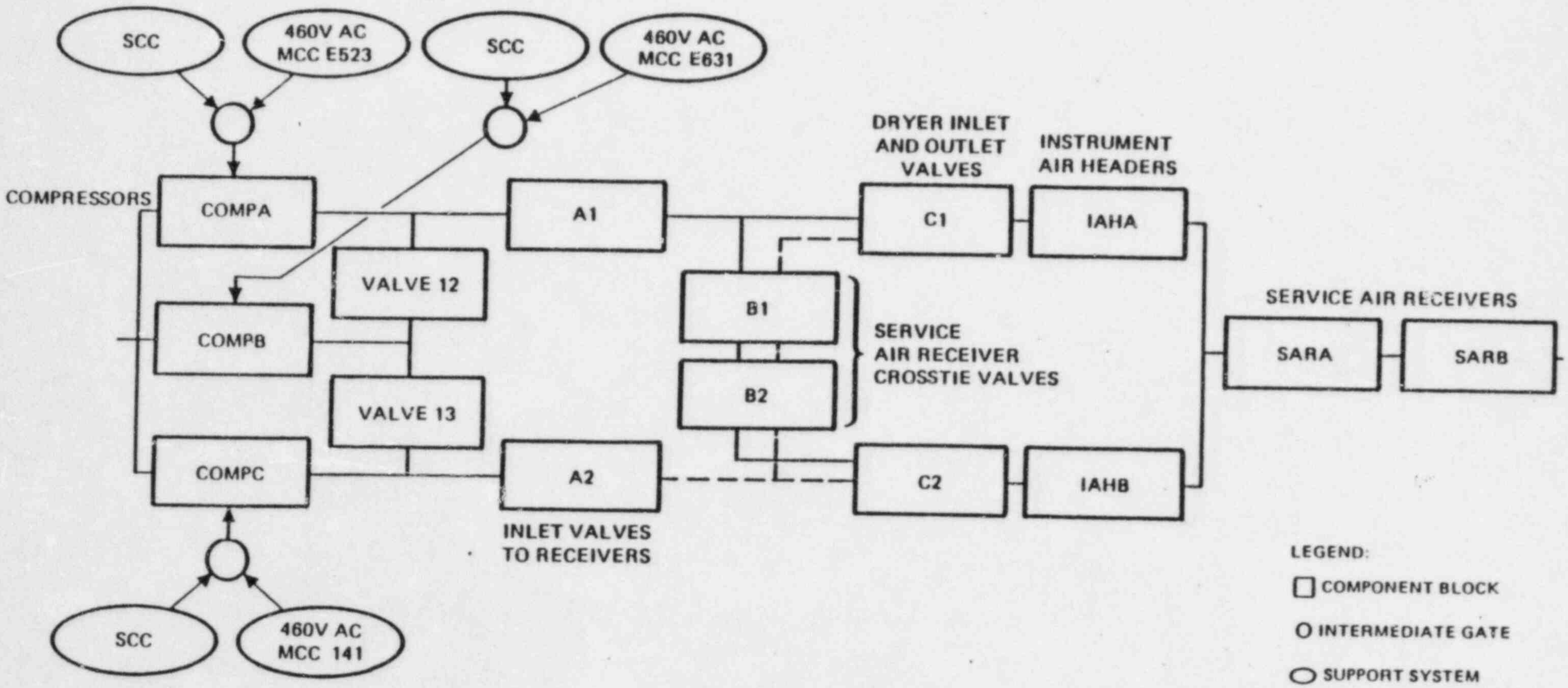


FIGURE 1. BLOCK DIAGRAM SHOWING SUPPORT SYSTEMS FOR INSTRUMENT AIR SYSTEM

ASSURING QUALITY IN RISK ASSESSMENT STUDIES

The essential differences between a rigorous scientific study and a nonscientific, intuitive evaluation are the use of appropriate and consistent methods, careful documentation, and peer review. PRA is a highly scientific endeavor requiring the highest levels of technical competence and integrity.

As with any scientific endeavor, the quality of a PRA study hinges on the use and documentation of appropriate assumptions, methods, data, and analysis. The purpose of careful documentation is essentially twofold. One major purpose is to aid the analysts in maintaining control over the process; i.e., it builds a "blueprint" of the progress which permits tracing logical progressions from initial assumption to final results. The second function of the documentation is to facilitate peer review, critiques, and reproducibility.

Given the requirements for a quality study, it is easy to see that the competence and integrity of the people involved are of paramount importance. For a PRA to be successful, the study team must be made up of at least the following:

- Experts in the analytical and probabilistic methods employed in the analysis.
- Engineers who have hands-on knowledge of the workings of the engineered systems being analyzed.
- Practitioners who can translate analytical methods and plant knowledge into meaningful models for quantifying risk.
- Engineers and scientists with concentrated knowledge of the behavior of systems under normal and abnormal conditions.
- Specialists in phenomena that are relevant to the study. Such phenomena might include earthquakes, fires, floods, and extreme winds.
- Authors who have special skills in communicating highly technical and scientific work.

The most important consideration for verifying the quality of a PRA is to perform the work correctly in the first place. Quality assurance is enhanced by segmenting the study into stages such that the analyst has checkpoints on his progress. Internal procedures require the analyst to present his work to his associates and defend the results. This technique is very effective in creating a sense of responsibility and professionalism. In addition, a different analyst checks the model and duplicates the key calculations. The work is subject to detailed review by senior members of the study team. This review team checks on the overall methods employed, makes spot checks of detailed models and calculations, questions all assumptions, carefully reviews all documentation, and identifies weakest and strongest points in the analysis. An external review board of technical experts provides an additional measure of quality assurance.

OBJECTIVES AND SCOPE OF THE SEABROOK STATION PRA

The objectives of the Seabrook Station Probabilistic Safety Assessment were to:

- Perform an independent and plant specific assessment of the health and safety risk to the public based on the unique factors associated with the site, design, plans for operations, maintenance, and emergency response at Seabrook Station.
- Provide documentation of the results and methods in a form suitable for detailed technical review as well as for presentation to the public.
- Provide a risk model which can be used to aid in decision-making with regard to possible modifications to the plant design, operations, and emergency response plans.

The Seabrook Station study is a full scope probabilistic risk assessment; i.e., it considers events that can initiate accidents from inside or outside the plant. Full scope also means that different levels and types of damage from potential accidents are analyzed and quantified. Of particular importance is the calculation of uncertainty for the different damage levels.

The most advanced state-of-the-art methods were used in the Seabrook Station risk assessment and, in some aspects, even more rigorous methods were employed than had been used in risk assessments of other nuclear plants. A key feature is that the study is plant and site specific; that is, the actual design and construction of Seabrook Station was analyzed. The results reflect the plant's geographic location and its relationship to the surrounding population.

The study and its results are an objective, independent assessment of the risks associated with the operation of Seabrook Station. The assessment provides information about the risks, but is not intended to determine or supplant social values.

QUALITY ASSURANCE OF THE SEABROOK STATION PRA

A full scope, plant specific PRA is a complex technical endeavor. To ensure a quality product, it is necessary that the study team be fully competent; that the work be carefully reviewed; and that the final documentation be complete in its reporting of all assumptions, calculations, data, methods, and results.

THE STUDY TEAM

The Seabrook Station PRA project was carried out by Pickard, Lowe, and Garrick, Inc. (PLG), as the main contractor. PLG managed the project, provided the methods and data base, and performed about 78% of the

technical work. PLG, its principals, employees, and consultants, are recognized as leaders in the development and application of risk assessment methods. A team of subcontractors selected by PLG supported them in the remaining 22% of the project. The subcontractors were selected for their particular expertise and competence in specific disciplines.

Structural Mechanics Associates, Inc. (SMA), provided an analysis of components and structures with respect to sensitivity to earthquake damage. They also performed analyses to determine the response of containment structures to accidents involving damage to the reactor core. Dames & Moore (D&M) performed the seismic hazard analysis which took into account previous assessments of seismic hazard at Seabrook Station performed by Weston Geophysical, Inc. Westinghouse Electric Corporation, as the reactor vender, provided information about the nuclear steam supply system.

Offshore Power Systems (OPS), a division of Westinghouse, performed analyses of the core and containment atmosphere for a wide spectrum of core damage scenarios. OPS also quantified the source terms and other release characteristics for the set of potential accident sequences relevant to the site (consequence) model. Fauske and Associates, Inc., provided site specific analyses of the behavior of a degraded core and debris formed within and outside the reactor vessel, which was also valuable input to the core and containment response analysis.

In support of the site analysis effort, Mesomet, Inc., provided data and analyses of the influence of the Atlantic Ocean on meteorological conditions, and Digital Graphics, Inc., supported post-processing of data generated by the computer-based site consequences model. The site analysis team also coordinated its efforts in modeling evacuation with HMM Associates, Inc., which is performing a separate study in support of the development of site emergency plans for Seabrook Station.

Several individuals who occupy distinguished faculty positions at prestigious institutions made important contributions to the project, acting as private consultants. Donald A. Norman, Director of the School of Cognitive Science at the University of California, San Diego, provided review and guidance in the modeling of human responses to accidents. Norman C. Rasmussen, Professor of Nuclear Engineering at the Massachusetts Institute of Technology, served on the technical review board for the project. C. Allin Cornell of Stanford University was an advisor on the seismic analysis.

To assure accuracy in the modeling and assumptions regarding "as-built" plant features, operating and maintenance procedures, and factors related to how the plant is to be managed, and to increase the usefulness of the risk model in support of decision-making, it was necessary to elicit input from the utility organizations responsible for startup and operation of Seabrook Station. These organizations are Public Service Company of New Hampshire (PSNH) and Yankee Atomic Electric Company (YAEC). Utility input to this project, which was coordinated through YAEC, consisted of providing information and documentation of the plant design, construction, and plans for operation.

QUALITY ASSURANCE PROCEDURES

The objective of the quality assurance (QA) program developed by PLG was to ensure that the services provided were reliable, traceable, and in full compliance with all applicable Federal regulations and industry standards. For this project, additional emphasis was placed on technical review. A description of the technical review levels is provided in Table 1. A brief description of the QA procedures follows:

- The document control system specifies procedures for identifying and logging documents transmitted and received and for storing and retrieving project files.
- Corrective action procedures establish requirements for controlling corrective actions for QA program deficiencies discovered during technical analysis and reviews or quality assurance program audits. The procedures address the responsibility for detection and correction of the deficiency, the filing of Corrective Action Reports (CAR), and the tracking of report status.
- Quality assurance program audit procedures establish guidelines for the frequency, scope, and documentation of internal audits, and the responsibilities of the company offices and managers. The internal audit is made to ascertain that the specified quality assurance procedures are being followed and to uncover any deficiencies in the procedures.
- Independent technical review guidelines establish the scope of the reviews and the responsibility of the project managers in these reviews.
- The computer code quality assurance program establishes the responsibilities of the project manager, computer coordinator, computer code author, and code verifier. The program also sets guidelines to ensure that the codes perform as intended and are properly documented.
- The document change control defines procedures for processing and approving changes to project documents. Project documents include the project plan, quality assurance manual, and any other documents affecting control of the project.
- Subcontractor selection procedures set responsibilities and selection and documentation guidelines to ensure that subcontractors meet the same technical and quality assurance standards as set forth in the manual.
- Federal regulation compliance procedures set guidelines to ensure that the appropriate lawful actions are taken should significant safety defects in the plant be revealed.

TABLE 1. REVIEW RESPONSIBILITIES

Stage	Review Objective	Person Responsible
1	Check all calculations, computer input and output; proofread documents prepared by publications department for technical accuracy.	Analyst/Author
2	Double-check all calculations; review documentation for technical accuracy; ensure consistency of documentation within technical area (e.g., systems); ensure that the right tools are used.	Task Leader
3	Spot check calculations; ensure that acceptable PRA methods and procedures are utilized; perform independent review of all deliverables and supporting calculations and documents as necessary focusing on reasonableness of results and conclusions and whether project documentation adequately reflects what was done; recommend corrective action when appropriate.	Technical Review Board
4	Review all deliverables; ensure Project objectives are met; ensure consistency among technical areas and documentation; responsible for resolution of all review comments and assignment of work needed to resolve review issues.	Project Manager
5	Assure that all parts of the project team perform their assigned responsibilities; review results and conclusions of key deliverables.	Project Director
6	Review all deliverables for correctness of interpretation of plant design and planned operation, documentation, safety analyses, and modeling of plant and site unique characteristics.	Client (PSNH and YAEC)
7	Perform QA audits; conduct QA training; maintain QA records.	PLG QA Manager

THE REVIEW PROGRAM

The scientists and engineers who performed the Seabrook Station study were chosen for their competence and integrity. To further ensure accuracy and thoroughness, PLG developed and implemented a review program that involved review by other qualified scientists, PLG management, and independent experts. The clients, PSNH and YAEC, reviewed PRA project documents for accuracy in modeling the plant and performed independent quality assurance audits of PLG and the subcontractors.

THE SEABROOK STATION MODELS AND ANALYSIS

The Seabrook Station risk assessment was divided into three major segments: the plant model, the containment model, and the site (consequences) model. The plant model segment is all of the modeling and analyses that trace potential accidents from initiating events up to a determination of the damage to the active plant systems. Several different accident sequences could lead to the same "plant damage state." Plant damage states are defined in terms of the time following core damage and the physical state of the safety systems. Therefore, the plant model also includes the work necessary to group scenarios into different plant damage states. This process enables further analysis to be done in a more efficient manner. The progression of an accident past the plant damage state is only dependent on the systems damaged and not on how that state was reached. For example, if the reactor core is uncovered early and the containment sprays have failed, the further analysis is the same whether this state was initiated by a loss of coolant accident (LOCA), a loss of offsite power, or some other circumstance.

The containment model consists of the work necessary to model the containment response to the various plant damage states. The containment model results in a set of release categories that identify the timing and quantity of radioactive releases from potential accidents.

The release categories from the containment model are the input to the site model. The site model traces the movement of radioactive releases, their fallout to the ground, and their effects on the population present. The damage predicted depends on the type of release, weather conditions at the time, the population pattern, and on whatever evacuation or sheltering actions are implemented.

THE PLANT MODEL

The objective of the plant model is to quantify the frequency of occurrence of different accident scenarios. The essential steps are:

1. Definition of a Comprehensive Set of Initiating Events. Initiating events that simultaneously fail safety systems that would otherwise terminate an accident sequence receive special treatment. Such

events are a special class of what is known as "common cause" events. Once a set of initiating events has been defined, they are grouped into categories to facilitate the structuring of accident scenarios.

2. Development of a Set of Event Sequences; i.e., Accident Scenarios Specific to the Seabrook Station. The initiating event categories from step 1 serve as input for structuring accident scenarios. Final disposition of the scenarios is dependent on the response of individual systems that could terminate or alter the sequence. Thus, this step involves extensive systems analysis including the processing of large amounts of reliability data. It also involves detailed analysis of system dependencies including the effect of equipment location and human error.

3. Quantification of the Accident Scenarios. The initiating event and system unavailability frequencies provide the input necessary to perform the quantification process. In effect, this information is propagated through the logic diagrams, that is, the event trees, to quantify the frequency of occurrence of the different plant damage states. The propagation includes the treatment of uncertainty.

The development of the Seabrook Station plant model required detailed modeling of the plant, its systems, components, and structures and all their interdependencies. To obtain the necessary knowledge of the plant, the PLG team first spent 6 weeks at the Seabrook Station site studying the plant. Information was obtained from physical inspection of the plant, engineering drawings, quality assurance documents, and discussions with the engineering consultants at YAEC. Additional visits were made by specialists to obtain information for expert analysis of such features as spatial relationships.

A detailed analysis was performed for the plant's systems. The analysis included a description of the system, its function and operations, development of a system logic model, and quantification of system unavailability. The systems analyzed in detail are identified in Table 2.

Particular care was taken to identify initiating events specific to the Seabrook plant and site. Six distinct approaches were used to identify possible initiating events including those from outside causes. Each approach had its own unique value in helping to identify an as-complete-as-possible set of potential accident initiators for Seabrook Station. The possible accident initiators include loss of coolant accidents; transients (a transient is a malfunction or error that impacts proper operation of equipment or systems); and common causes such as fires, turbine missiles, tornados, hazardous chemicals, loss of support systems, and seismic events.

The sequence of events that could follow each initiating event was defined using event trees. Literally billions of potential scenarios were developed and analyzed. Data such as component failure rates and

TABLE 2. SEABROOK STATION SYSTEMS ANALYZED IN DETAIL

Electric Power System
Service Water System
Primary Component Cooling Water System
Instrument Air System
Reactor Trip, Engineered Safety Features Actuation and Solid State Logic Protection Systems
Containment Enclosure Air Handling System
Emergency Core Cooling System
Emergency, Main, and Startup Feedwater Systems
Reactor Coolant Pressure Relief
Main Steam System
Containment Building Spray System
Containment Isolation Functions
Control Room Complex Heating, Ventilation, and Air Conditioning

component maintenance frequency and duration were obtained from industry data bases to quantify the frequencies of the scenarios. These data were derived from experience with operating the same type of systems and components at other plants. Expert opinion and mathematical techniques were used for initiating events or components for which data is sparse.

The modeling and analyses incorporated such design features as the condenser cooling water system; the service water system (which also depends on seawater); Seabrook Station's advanced control room alarm indicators; and environmental factors unique to the plant's site such as the potential for earthquakes, tornados, floods, aircraft crashes, and fires.

The results of the plant model, a listing of groups of possible damage states to the plant, formed the input for the containment model.

THE CONTAINMENT MODEL

The containment model consists of analyses of degraded reactor cores, core melt processes, and thermodynamic conditions in the containment building; an assessment of the ability of the containment structure to withstand these thermodynamic conditions; and radioactive release analyses. The starting point of the containment model is the set of plant damage states identified in the plant model. Given that a particular plant damage state or external event has occurred, the subsequent events are represented by a containment event tree. Each path through the containment event tree begins with a plant damage state and ends with a "release category." A release category represents the types, quantities, timing, and elevations of radioactive material released, if any.

The core and containment response analysis of Seabrook Station made maximum use of the same research and development programs used for the Indian Point risk assessment. Specifically, the core and containment response analysis performed for the Indian Point Units 2 and 3 risk assessment could be applied to Seabrook for several reasons. First, there is the strong similarity in design and construction of the Indian Point and Seabrook plants. The Indian Point PRA was performed by the same team of analysts. The very extensive core and containment response analyses performed for the Indian Point study have undergone detailed review and acceptance by the Nuclear Regulatory Commission. While the methodology and models used for Indian Point are directly applicable to Seabrook Station, considerable care was taken to address differences in the design and construction of the two plants. For example, Seabrook Station's "double containment" (i.e., the existence of a containment enclosure building outside the main containment) is a significant departure from the Indian Point design.

During accident sequences involving core damage, the containment structure can be exposed to pressure and temperature conditions which

exceed the design basis for the containment. In order to quantify the time, rate, and magnitude of radioactive releases, it is necessary to know the following:

- The internal pressure at which the containment is realistically expected to fail.
- The location of the failure.
- The leakage path and the effective leak rate.
- The condition of the enclosure building when the containment building fails.

The Seabrook Station containment model examined the different ways the containment building could fail during each important accident scenario.

THE SITE MODEL

Given that a particular release has occurred including the magnitude and timing of the radionuclide mix, the site model is used to determine the consequent damage to the area and people around the plant. For each particular release, this model traces the movement of the radioisotopes, their fallout to the ground, and their interaction with the population present given a variety of weather conditions. The resulting damage calculated depends not only on the weather conditions (wind direction, speed, precipitation, etc.), but also upon the population pattern and the evacuation or sheltering actions implemented. Several different measures of damage are customarily used to present the risk results: early deaths, early injuries, thyroid cancers and other cancer fatalities (latent), whole body dose, property damage, etc.

An elaborate computer program is used to model radioactive releases and their interaction with the specific topography, meteorology, and demography relevant to Seabrook Station. Meteorological data for the Seabrook site include air flow patterns, storms producing rain or snow, land-sea interactions, and seasonal variations in the weather. The terrain around the Seabrook site is relatively flat. Previous analyses of sites with flat terrains have shown that local topography does not significantly influence local dispersion of radioactive material. Therefore, no terrain effects on dispersion were considered for Seabrook Station.

Three sets of population and evacuation data were prepared for an evacuation zone consisting of the area within 10 miles of the plant. The three scenarios used are the winter weekday, summer weekday, and summer weekend day. The last of these represents the worst case scenario. An evacuation model for each of the three scenarios was developed by HMM Associates. The model included traffic routes, travel distances, and delay times. The biological effects of radiation exposure were calculated in accordance with methods used by the National Academy of Sciences in the report entitled, "The Biological Effects of Ionizing Radiation" (BEIR) (Reference 1).

ASSEMBLY OF RISK MODELS

The final step in a risk assessment is the assembly of the individual risk models into a full statement of risk on the Seabrook Station. The four components, or "pinch points" of a full scope risk assessment have been identified as: (1) initiating events; (2) plant damage states; (3) release categories; and (4) final damage states (consequences). The last three pinch points are a direct result of the three models covering the plant, its containment, and the site. While the pinch points are sequentially dependent, the methodology permits them to be developed independently until the final assembly step. Thus, the unconditional results from pinch point 4 depend on the results of pinch point 3 and so on to pinch point 1. The assembly process removes the dependencies on each of the three models.

RISK ASSESSMENT RESULTS

KEY FINDINGS

The risk to the public from the operation of Seabrook Station is presented in terms of the frequency of occurrence of different levels of damage; in this case, different types of health effects. Five different offsite health effects were considered: (1) early fatalities; i.e., fatalities occurring within a short time after radiation exposure; (2) nonfatal radiation injuries due to exposure; (3) thyroid cancers (most of which are treatable and nonfatal); (4) latent cancer fatalities occurring over a 30-year period; and (5) total population dose or man-rem (whole body). For purposes of highlighting the health and safety risk of Seabrook Station, three of these indices are discussed: (1) early fatalities, (2) early radiation injuries, and (3) latent cancer fatalities. In addition, the frequency of core melt is quantified. It is necessary for the core to melt before significant amounts of radioactive material can be released into the containment and, hence, the interest in core melt.

EARLY FATALITIES

It is clear from the Seabrook Station Probabilistic Safety Assessment that events leading to fatalities due to exposure to radioactive material following an accident are indeed very rare. An "upper bound" estimate* on the frequency of events that result in a small number of fatalities, say 1 to 100, is one such event every half million to a million years. The "best" estimate for the frequency of such an event is one every

*Upper bound is taken to be the 95% confidence value which has the meaning that we are 95% confident that the indicated frequency of occurrence of the damage level in question will not be exceeded. The best estimate is the 50% or median value.

30 million to 50 million years. Thus, for all practical purposes, there is no appreciable risk of early fatalities from the operation of Seabrook Station. The reason for essentially no early fatality risk is related to the very high strength of the Seabrook Station containment. The ultimate strength was analyzed to be above 170 pounds per square inch (gauge) --nearly three times the design pressure. Thus, containment failure is almost an impossibility. About the only accidents that directly fail the containment and contribute to risk are those that occur some 2-1/2 days or so following a damaged core. Such accidents must result in a loss of all containment heat removal capability. As is observed below, these accidents affect, for the most part, only the delayed health effects.

The only failure mechanism that has any appreciable chance of leading to fatalities immediately (within hours) following an accident is one that would either permit a bypassing of the containment system through coolant pathways between the reactor coolant system and systems outside the containment; or, even less likely, an earthquake of such severity as to simultaneously fail the containment and lead to core meltdown. Of course, containment bypass has been essentially "designed out" and earthquakes strong enough to fail the containment and cause a meltdown have never been observed or diagnosed in the vicinity of Seabrook.

LATENT CANCER FATALITIES

In addition to the health effects that occur near the time of an accident, there is the need to consider residual or latent effects. Thus, consideration is given to cancer fatalities occurring over a 30-year period following exposure. The frequency of events that can contribute to cancer fatalities is much greater than those leading to early effects. For example, those releases that develop slowly (say 20 hours or more following an accident) can contribute to latent cancers but most likely would not result in any early fatalities. One reason for this is that ample time exists to remove people from the zones susceptible to lethal doses of radioactive material. Therefore, radiation exposures to individuals would fall below thresholds for producing early health effects. Another factor is that when a delay is involved, there is more time for radioactive material to settle out inside the containment as well as to decay--both contributing to reductions in the inventory of material available for release.

The upper bound estimate of the frequency of occurrence of accidents resulting in 1 to 100 cancer fatalities is once every 3,000 to 5,000 years. The best estimate frequency is once in 7,000 to 20,000 years. While even less frequent than events that lead to early fatalities, these frequencies are still very small especially when considering the frequencies associated with other risks routinely taken. Even though the latent fatality risk is small, it is important to know what is contributing to it to enable actions to keep it small or make it even smaller. As already implied, the events most responsible for the latent effects are those involving delayed releases; that is, a delay between when fuel damage occurs and when a release actually takes place. What this means is that at Seabrook Station, there is containment of the damaged fuel until well after such time at which all core cooling and containment heat removal systems fail. Without heat removal capability,

the containment integrity cannot be assured indefinitely. Of course, such a combination of events is very unlikely, as the low frequencies indicate.

EARLY INJURIES

Early injuries include radiation illnesses that are usually observed after large, acute doses of radiation. They can occur within days to weeks after exposure. Such injuries also include illnesses that are manifested within a year or more following exposure. Thus, early injuries involve those radiation illnesses that result from high exposures occurring at or near the time of a very major accident and that generally do not result in fatalities. For Seabrook Station, the risk of such exposures is extremely small. The upper bound of the frequency of events leading to 1 to 100 early injuries is calculated to be once every 200,000 to 300,000 years. The best estimate of this frequency is once every 7,000,000 to 17,000,000 years.

These results again indicate that the risk of high radiation exposure events at Seabrook Station is essentially zero. Since both early fatalities and early injuries require large, acute doses of radiation (the kind that can only come from especially severe accidents), they are both caused by the same types of initiating events. Thus, the principal contributors to early injuries are events that somehow bypass the containment.

The above results of the risk assessment have broken out the separate components of accident likelihood, as measured by its frequency, and consequences as measured by early and latent fatalities and injuries. A similar perspective of these results is afforded by combining the frequency and consequence values into a single risk measure. This is done to express risk in terms of the expected frequency of health effects projected for the population near Seabrook Station. The tabulation below shows the minute increase in risk estimated for the period when Seabrook Station is in operation.

Health Effect	Population Segment	Risk (number of health effects per thousand people per year)	
		Before Plant Operation (nonnuclear causes)	During Plant Operation (all causes)
Early Fatality	4,435 people within 1 mile of Seabrook Station.	0.5	0.5002
Latent Cancer Fatality	4,200,000 people within 50 miles of Seabrook Station.	2.0	2.00001

CORE MELT

In order for there to be any offsite radiation risk from a nuclear plant, it is necessary that the fuel be severely damaged and the containment integrity violated. Thus, a necessary but not sufficient precursor event to any adverse health effects from a potential accident at Seabrook Station is core damage or core melt. Core damage does not always lead to offsite health effects. In fact, as long as containment integrity is maintained, there will not be a significant release no matter what happens to the core. Such was the case during the Three Mile Island accident. Thus, the frequency of events leading to offsite injuries and fatalities following an accident is, as expected, far less than the frequency of events damaging the core. That is, it is not just a case of core damage but how the core damage takes place that dictates the outcome of an accident. For example, core melts that incapacitate containment systems are certainly much more serious and less frequent than those that do not.

The best estimate of the core melt frequency for Seabrook Station is calculated to be 1 in 5,300 per reactor year with an upper bound of once every 2,400 reactor years. While the frequency of occurrence of core melt is considerably greater than that of injury or fatality, it can still be considered a rare event. Events which contribute most to the core melt frequency involve failures in electric power to operate important safety systems. Other contributors include such external events as very large earthquakes and fires and, to a lesser extent, accidents that are caused by a loss of coolant. Of course, these are all events that occur very infrequently; thus the small risk of core damage.

RISK AND UNCERTAINTY

The above results have been given for the most part in terms of point estimates and ranges. It is important to try to put these results into their proper context. In the Seabrook Station Probabilistic Safety Assessment, great care was taken to quantify and communicate the uncertainty associated with the results. The language for communicating uncertainty is probability. In particular, for each damage index such as injuries, a family of probability curves was generated on the frequency of occurrence of different levels of damage. Those "risk curves" are in what is referred to as risk assessment in "probability of frequency" format. For example, the following figure shows the risk curves for radiation injuries generated for Seabrook Station.

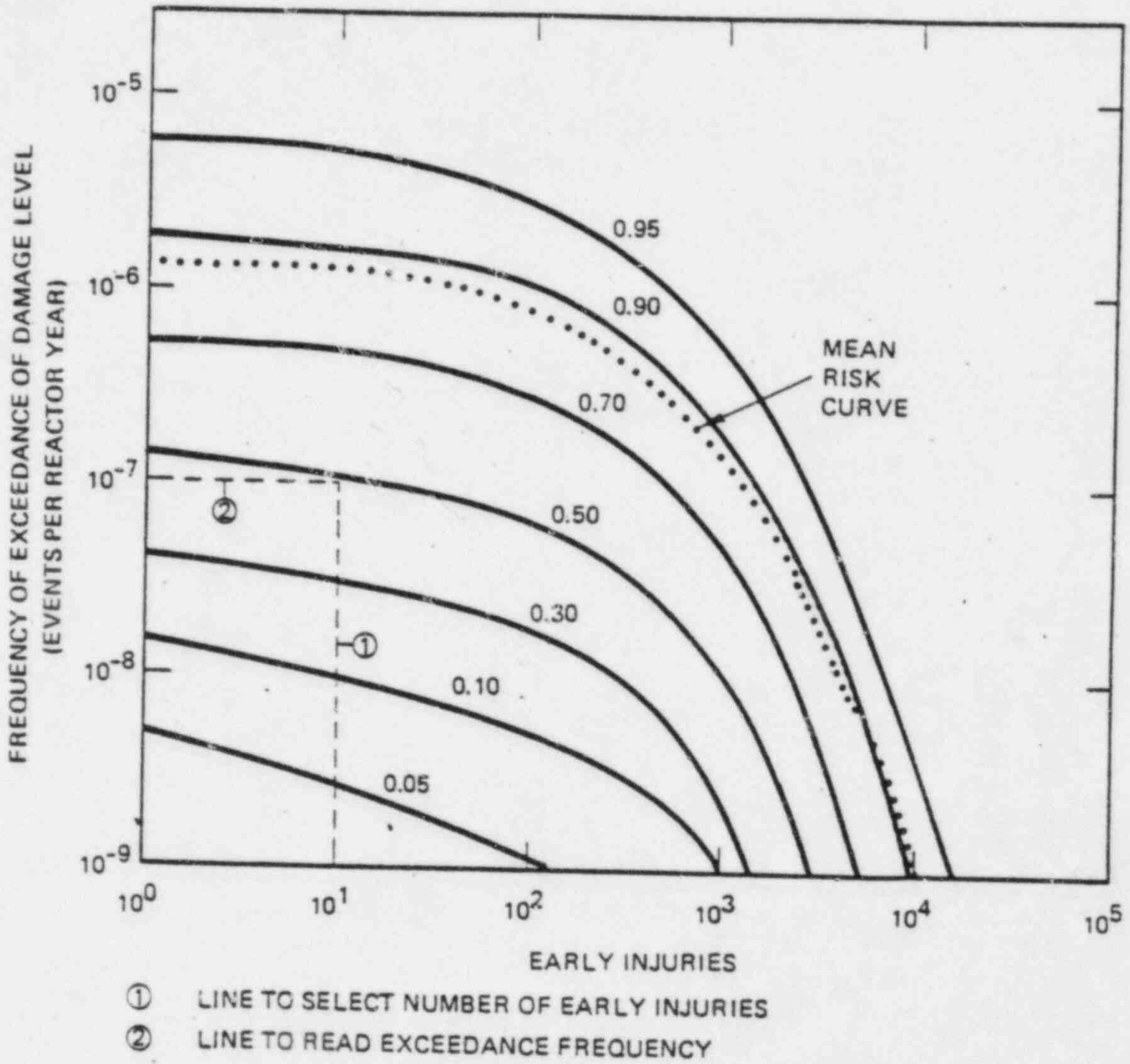


FIGURE 2. RISK OF RADIATION INJURIES AT SEABROOK STATION

Suppose it is desired to know the best estimate of the frequency of having 10 or more injuries from an accident at Seabrook Station. If we take as the best estimate a probability of 0.5 (that is, the 50% confidence value), we see that the frequency is 10^{-7} per reactor year. Inverting this frequency indicates that the damage level of 10 injuries occurs but once every 10,000,000 reactor years. Curves of this type are the source of the results presented in the section entitled "Key Findings." Table 3 presents some additional results from the SSPSA.

PERSPECTIVES

In general, the health and safety risk from Seabrook Station is extremely low, far lower than other risks to which we are routinely exposed. While comprehensive risk studies such as the SSPSA have not been made on

TABLE 3. THE RISK AT SEABROOK STATION

Best Estimate (50% probability) Frequency of Exceeding Number of Indicated Early Fatalities (per reactor year)	Number of Early Fatalities	Upper Bound (95% probability) Frequency of Exceeding Number of Indicated Early Fatalities (per reactor year)
3.1×10^{-8} or 1 in 32,300,000	1	1.7×10^{-6} or 1 in 590,000
2.7×10^{-8} or 1 in 37,000,000	10	1.4×10^{-6} or 1 in 710,000
2.1×10^{-8} or 1 in 47,600,000	100	1.1×10^{-6} or 1 in 910,000
1.2×10^{-8} or 1 in 83,300,000	1,000	6.9×10^{-7} or 1 in 1,500,000

Best Estimate (50% probability) Frequency of Exceeding Number of Indicated Latent Cancer Fatalities (per reactor year)	Number of Latent Cancer Fatalities	Upper Bound (95% probability) Frequency of Exceeding Number of Indicated Latent Cancer Fatalities (per reactor year)
1.4×10^{-4} or 1 in 7,140	1	3.8×10^{-4} or 1 in 2,630
1.1×10^{-4} or 1 in 9,100	10	3.1×10^{-4} or 1 in 3,230
5.4×10^{-5} or 1 in 18,500	100	1.8×10^{-4} or 1 in 5,560
7.4×10^{-6} or 1 in 135,000	1,000	4.4×10^{-5} or 1 in 22,700

Best Estimate (50% probability) Frequency of Exceeding Number of Indicated Radiation Injuries (per reactor year)	Number of Radiation Injuries	Upper Bound (95% probability) Frequency of Exceeding Number of Indicated Radiation Injuries (per reactor year)
1.4×10^{-7} or 1 in 7,140,000	1	5.7×10^{-6} or 1 in 175,000
1.1×10^{-7} or 1 in 9,100,000	10	4.8×10^{-6} or 1 in 208,000
5.8×10^{-8} or 1 in 17,200,000	100	3.0×10^{-6} or 1 in 330,000
9.3×10^{-9} or 1 in 107,500,000	1,000	4.7×10^{-7} or 1 in 2,130,000

alternative energy sources, there are some qualitative estimates. Based on one study (Reference 2), there is reason to believe that greater risk in terms of accidental death rates would result if Seabrook Station were replaced by a fossil plant (other than one fueled by natural gas). Thus, not only has the risk been assessed to be extremely small, but among the alternatives available there are indications that Seabrook Station is the most attractive alternative with respect to health and safety.

There are other perspectives from which to consider the SSPSA results. One of these is the safety goals covering nuclear power plants proposed by the NRC. The NRC safety goals have been published for a 2-year trial period to supplement already existing guides and regulations but not to replace them. There are two aspects of the goals. The first has to do with the societal risk and the second relates to the risk to an individual. Societal risk considers the cancer fatalities in the population within a 50-mile radius of the power plant that might result from its operation. Individual risk involves the early (acute) fatalities within 1 mile from the plant that might result from a nuclear plant accident. The societal risk calculated in the SSPSA is only a very small fraction of the safety goal when applied to Seabrook Station. In particular, the calculated risk at Seabrook Station is only one-thousandth to one-hundredth of the safety goal. The reason for the range is to reflect the uncertainty associated with such calculations.

With respect to the safety goal for individual risk, again the calculated risk for Seabrook Station is well below the applicable safety goal. Specifically, the risk of early fatalities to the population within 1-mile of the plant was found to be between a factor of 5 and 6 below the individual risk goal.

Reflecting on the Seabrook risk analysis results and how they compare with other risks and the NRC safety goals, in one respect, leads to somewhat of a letdown. The reaction is "what is all the fuss about?" Here is a situation where consideration has been given to billions of different accident sequences, millions of pieces of data, hundreds of thousands of dollars worth of computer runs, dozens of logic models, tens of thousands of calculations, and what does it tell us? It tells us that, for all practical purposes, there is no discernable risk to the health and safety of the public. These are the most sophisticated and comprehensive risk assessments ever performed on any system of any kind. Surely we must be able to get more out of it than risk--a risk very much invisible against the risks routinely taken by all members of the public. In fact, we do get more out of it than that.

First, we learn that the risk is indeed extremely small based on the most thorough and systematic analysis techniques at our disposal (and now, based on nearly 1,500 plant years of commercial reactor experience throughout the world). Second, we have a comprehensive risk model of the plant--a model fully capable of being employed as a management tool. Thus, the opportunity for controlling risk and keeping it small is greatly enhanced. We know more about what to expect when certain things go wrong, what the alternatives are for corrective action, and how to

quantify the effect on risk of changes in the plant. Additionally, we are in a position to include such knowledge in training and in creating a greater awareness and respect for the responsibility of safe operations and effective emergency response should that rare event actually and surprisingly occur.

By tackling the extremely difficult problem of quantifying the frequency of the very rare events that threaten public safety, we bring ourselves much closer to better controlling more frequent events--events that, while not a safety threat, are nevertheless very important to making power generation a much more cost effective and attractive business. This occurs because the same principles can be applied to quantify financial risk events as are used to quantify health and safety risk. Such management enhancement benefits all.

REFERENCES

1. National Academy of Sciences, Committee on the Biological Effects of Ionizing Radiations, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," 1980.

2. National Academy of Sciences, Energy in Transition 1985-2010, W. H. Freeman and Company, San Francisco, 1980, p. 429.